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BASIC RADIOLOGICAL HEALTH

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Sources of Radiation Exposure

I. SOURCES OF RADIATION EXPOSURE

The human race has always been exposed to ionizing radiation of cosmic origin and from natural sources in the environment and within the body. A representative distribution of the dose received from this source is shown in Table I.(1) The indicated total of 125 millirems per year is an estimated average figure for the United States. The exact amount varies with locality because of differences in the amounts of natural radioactive material present and variation in the intensity of cosmic rays with altitude and latitude. For example, cosmic radiation increases by a factor of 2 in going from sea level to 10,000 feet and by about 15% in going from 0° to 50° geomagnetic latitude. Reported natural background radiation levels for the United States range from about 90 to 200 millirems per year. In some parts of the world, such as the Kerala regions of India and the States of Minas Gerias and Espirito Santo of Brazil, much higher levels prevail.(2) Radiation from thorium-bearing monazite sands in these areas makes the external environmental absorbed dose 10 to 30 times the world average. Man has reached his present state of development in the presence of background radiation; it is apparently an amount with which he can live compatibly.

TABLE I.--Representative Distribution of Natural Background Radiation

Source	Dose (mrem/yr)	
	Gonad	Bone Marrow
External		
Cosmic Rays	50	50
Terrestrial*	50	50
Internal		
Potassium	20	15
Radium	1.3	1.6
Lead	0.3	0.4
Carbon	0.7	1.6
Radon	3.0	3.0
Total	125	122

*This is due to the presence in the earth's crust of natural radioactive elements, principally those associated with the uranium, thorium, and actinium decay series and with potassium.

Today, background radiation is only one of many sources to which the world's population is exposed. Radiation-generating machines and radioactive materials present in many phases of the environment constitute the principal manmade sources.

II. X RAYS

X-ray machines are finding widespread use in industry, medicine, commerce, and research. All such uses are potential sources of exposure.

A. Medical and Dental Use

Currently, over 200,000 medical and dental x-ray units are used in the United States for diagnosis and therapy. Some 98,000 of these units are used by dentists, with 100,000 to 130,000 employed by medical practitioners, radiologists, hospitals, and clinics.(3)

Surveys indicate that a substantial number of medical-technical personnel are occupationally exposed in the operation of these machines. In addition, a considerable portion of the general population is exposed during diagnostic and therapeutic procedures. On the average, over half the people in the

United States are exposed to x rays for medical or dental purposes each year. In 1964, the number of persons x rayed(4) was approximately 108 million--12 million of whom had more than one category of x-ray procedure which results in the following distribution:

66 million for radiographic procedures
46 million for dental diagnosis
8 million for fluoroscopy
600 thousand for x-ray therapy

B. Industrial Use

Industrial x-ray devices include primarily: (1) radiographic and fluoroscopic units used to determine defects in castings, fabricated structures and welds, and (2) fluoroscopic units used to find foreign material in such items as food products. At present, there are slightly more than 10,000 industrial radiographic installations in the United States.(3)

C. Commercial Use

A major use of x-ray machines has been in shoe-fitting. In 1950, there were almost 10,000 such units in operation in the United States.(5) Health authorities considered such exposures unnecessary, and the application of x-ray fluoroscopes in shoe-fitting has been banned or severely restricted in most States. The number remaining in operation today is small.

D. Research Use

High-voltage x-ray machines and particle accelerators are familiar features of research laboratories in universities and similar institutions. In 1941, there were only 16 cyclotron laboratories in the United States. Today, well over 600 accelerator installations (which include cyclotrons, synchrotrons, van de Graaff generators, and betatrons) are in operation.(3,6) Other x-ray equipment used in research includes diffraction units. Over 4,000 of these units are currently in use.

An emerging source of manmade, machine-produced radiation is the neutron generator. In 1966, there were over 200 of these with about 50 additional units being sold annually. Cathode-ray tubes, high-voltage electronic rectifiers, transmitting valves, and television and image tubes can also be significant sources of x-ray exposure if voltages are high and protection is inadequate.

III. RADIONUCLIDES

A. Naturally Occurring

The naturally occurring radionuclide most widely used in medicine and industry is ^{226}Ra. In considering the health aspects of naturally occurring radionuclides such as ^{226}Ra, it should be noted that this radionuclide is not covered by AEC regulations and, in some states, can essentially be purchased on the open market. Recently some states have enacted legislation on registration and licensing of naturally occurring radionuclides.

The history of the application of radium and its daughter product is replete with instances of injury and death. This is particularly true in the radium dial painting industry and in mining operations. Even today, radium continues to be a problem, especially in its medical applications. A recent survey in Georgia(7) revealed that of 210 hospitals contacted, 24 owned or leased radium and 40 more allowed its use in their facilities. For the 24 hospitals, the amount of radium on hand ranged from 50 to 440 milligrams (millicuries) and the numbers of treatments ranged from 1 to 100 per year. On-site surveys showed that eight of the 24 hospitals had one or more leaking sources. Only five routinely performed leak tests.

The national inventory of radium has been estimated at 2,000 grams. About 700 grams can be accounted for as sealed sources. Of the remaining 1300 grams, perhaps 65 to 70 percent has gone into various luminous compounds for use on wrist watches, aircraft, military instruments, etc. The 700 grams that are identifiable are used in approximately 4,500 facilities: approximately

1,000 hospitals and clinics, 1,700 private practitioners' offices, and about 1,800 industrial research and educational institutions.(8)

B. Artificially Produced

Radionuclides can be produced on a practical scale by neutron irradiation in reactors, by recovery from fission wastes, or by bombardment of target materials by charged particles in accelerators.

The United States has tremendous reactor production capacity. Large amounts of cesium-137, strontium-90, promethium-147, and other fission products are present in fission wastes at the United States Atomic Energy Commission's fissionable materials production sites. The Commission has announced that megacurie amounts of a variety of radioisotopes are available for purchase.(9) (See Tables 2 and 3.)(10)

Over 11,000 universities, hospitals, and research laboratories in the United States are using manmade radionuclides for medical, biological, industrial, agricultural, and scientific research and for medical diagnosis and therapy.(3, 11) The Atomic Energy Commission and various state health departments have issued over 3,700 licenses to the medical profession.(3,11) These licenses authorize medical institutions and private physicians to use radionuclides for the diagnosis and treatment of illness. Over a million people in the United States receive this treatment each year.

Harmful exposure from such radionuclides is possible in their preparation, handling and application, and transportation. Exposures, internal and external, might also arise through contamination of the environment by wastes originating from the use of these materials.

IV. NUCLEAR REACTOR OPERATIONS

Sources of radiation exposure associated with nuclear reactor operations include the reactor itself, its ventilation and cooling wastes, procedures associated with the removal and reprocessing of its "spent" fuel and the resulting fission product wastes, and procedures associated with the mining, milling, and fabrication of new fuels.

There are nearly thirty power reactors, with approximately ten million kilowatts capacity, now operable or under construction in the United States. An additional 45 nuclear power plants with 33 million kilowatts are planned to be operational by 1976.(12)

The most extensive military application of reactors is for propulsion of Navy ships. At the end of 1967, there were 79 nuclear-powered ships in the active fleet: 75 submarines and four surface vessels (the aircraft carrier Enterprise, the cruiser Long Beach, and the guided missile frigates Bainbridge and Truxton). Under construction were 29 additional submarines and negotiations were underway for construction of another nuclear aircraft carrier. Some vessels are powered by more than one reactor. (For example, the aircraft carrier Enterprise has eight nuclear reactors.)(13, 14)

The Navy has also launched a series of TRANSIT navigation satellites powered by plutonium-238 thermoelectric generators.(15) It has also placed a series of nuclear-powered signal-producing devices on the ocean floor in various parts of the world.

Much work has also gone into the theory of atomic-powered flight and space systems development employing isotopic heat and power applications. In addition, there will be the need for more land-based research facilities and reactor-powered research propulsion units.

Many workers are currently engaged in uranium mining and milling operations in this country, primarily in the New Mexico and Colorado Plateau regions. The number of workers involved in the fabrication of new fuels and reprocessing of "spent" fuels is currently on the increase. The predicted activity in terms of curies of fission product wastes from such operations is extensive. In addition to this occupational exposure, the general population is potentially subject to low-level internal and external radiation

TABLE 2.--Total Distribution (Ci/yr) of Radioactive Material

	Estimated Volume		Potential Volume	
	1962	1970	1962	1970
Tritium	13,025	16,025	213,000	217,000
Krypton	52	8,501	52	8,501
Promethium	50	--	50	200
Polonium	1,750	2,500	1,750	2,500
Others	--	--	--	--
Total	14,877	27,026	214,852	228,201

TABLE 3.--Product Use (Ci/yr) of Radioactive Material

Use	Estimated Volume		Potential Volume	
	1962	1970	1962	1970
Direct Purchase Products				
Air ionizers	2	2	125,000	125,000
Portable air ionizers	23	23	75,000	75,000
Anti-static devices	1,750	2,500	1,750	2,500
Auto altimeters	50	--	50	200
Educational kits	<0.2	<0.3	<0.2	<0.3
Total	1,825	2,525	201,800	202,700
Continuous Exposure Products				
Aircraft signs	12,000	16,000	12,000	16,000
Railroad signals	51	8,500	51	8,500
Watershed tracers	1,000	--	1,000	1,000
Discriminator tubes	1	1	1	1
Iron ore tracers	--	--	--	--
Petroleum and pipeline interface tracers	--	--	--	--
Total	13,052	24,501	13,052	25,501

exposure associated with the possible escape and release of reactor wastes to the environment.

V. WEAPON TESTING

The worldwide total of nuclear detonations through 1962 is probably over 300. More important than the number of detonations is their energy yield, an indication of the quantity of fission products produced. Information released at the 1959 Congressional "Fallout Hearings" showed that 90 to 92 megaton fission yield equivalent had been released to that date. Table 4 [16] summarizes the fission and total yields of atmospheric tests conducted by all nations through 1962. Some radioactive materials from these detonations remain for a time in the upper atmosphere. However, measurable amounts of fallout are widely distributed over the United States, causing a temporary average increase in radioactivity throughout the country.

VI. POPULATION DOSE

In expressing radiation exposure, one should specify the part of the body being considered. Because of genetic effects, many people consider the dose to the gonads or reproductive organs of prime importance. Typical exposures from natural background and manmade sources are summarized in Tables 1 and 5. [1, 17]

VII. CONTROL PROCEDURES

With increasing knowledge and experience in radiological health, measures for

TABLE 4.--Approximate Yields (in Megatons) of Atmospheric Nuclear Tests Conducted by All Nations

Inclusive years	Fission Yield		Total Yield	
	Air	Surface	Air	Surface
1945 to 1951 . . .	0.19	0.52	0.19	0.57
1952 to 1954. . . .	1	37	1	59
1955 to 1956. . . .	5.6	7.5	11	17
1957 to 1958	31	9	57	28
Subtotal. . . .	37.8	54	69.2	104.6
1959 to 1960. . . .	*	*	*	*
1961.	25 †		120	
Subtotal. . . .	63	54	189	105
1962.	76 †		217	
Total	139	54	406	105

* Test moratorium.

† The small-yield tests conducted in Nevada do not contribute significantly to the worldwide distribution of strontium-90 to which this summary is related.

minimizing radiation exposures have been developed.

A. X-ray Control

Control procedures for protecting the operators of medical, dental, and industrial x-ray machines consist of limiting the time of exposure, maintaining an adequate distance between the x-ray beam and the operator, and using adequate shielding. Generally, a combination of these three safety measures can restrict the dose to acceptable levels.

In protecting the patient, two important factors are: limitation of the size of the beam through proper coning or collimation, and filtration of the beam to remove soft x rays which would otherwise be absorbed by the body. Experience has shown that the use of proper filtration can result in a two- to fivefold reduction in the dose to the reproductive organs of the patient; through proper coning, the genetic dose can be reduced by a factor of 200 to 500.[8]

B. Radioactive Materials

For radioactive materials, containment and isolation or dilution and dispersion are the main control procedures. Containment and isolation is the method of choice for low-volume high-level wastes; dilution and dispersion is used for large-volume low-level wastes. Intermediate-level wastes are generally handled by a combination of the two approaches; the radioactive material content is concentrated and confined with the decontaminated residue released to the environment. The chemical nature of certain radionuclides makes this combined approach difficult to apply. Examples of intermediate-level wastes are hydrogen-3 and krypton-85—two of the contaminants produced in nuclear reactor operations and fuel reprocessing plants.

VIII. COMMENTARY

For the United States, estimates indicate that medical and dental x rays contribute about one-third of the current genetic dose to man; natural background accounts for approximately 60 percent; and less than 10 percent arises from nuclear reactor operations, weapons testing fallout, and other manmade sources. When planning control procedures to minimize such exposures, it is important to consider the relationship between the dose and the resulting biological effects.

Such considerations are also uppermost in the minds of those responsible for the establishment of Radiation Protection Guides.[17] Most toxic materials, for example, have a threshold dose which, it is felt, must be exceeded before effects appear. When this is the case, control procedures are straightforward and the primary objective is to keep the dose below the threshold. With ionizing radiation, however, it is believed that even the smallest dose may be capable of producing harmful genetic effects. The same relationship may hold for somatic effects.

Since radiation damage may have no threshold, the philosophy of radiation control generally reflects a weighing of the benefits of the particular radiation application against the risks of the associated exposure. That is, the dose must always be kept below the level at which the harmful effects would be considered to outweigh the benefits.

Based on these considerations, experts[17] have made recommendations for the safe conduct of radiation operations. Guide levels have also been established for the general population. These provide criteria for minimizing genetic and somatic damage. For health personnel, the challenge is to be familiar with exposure sources, to understand the effects of exposure upon man, and to establish effective control programs which will keep occupational and environmental exposures at acceptable levels.

The increasing magnitude of radiation-producing sources is shown in Figure 1. Simultaneous with this increase, the recommended maximum permissible levels for radiation exposure, now called Radiation Protection Guides,[17] have decreased sharply. The magnitude and sequence of this decrease is seen in Figure 2.

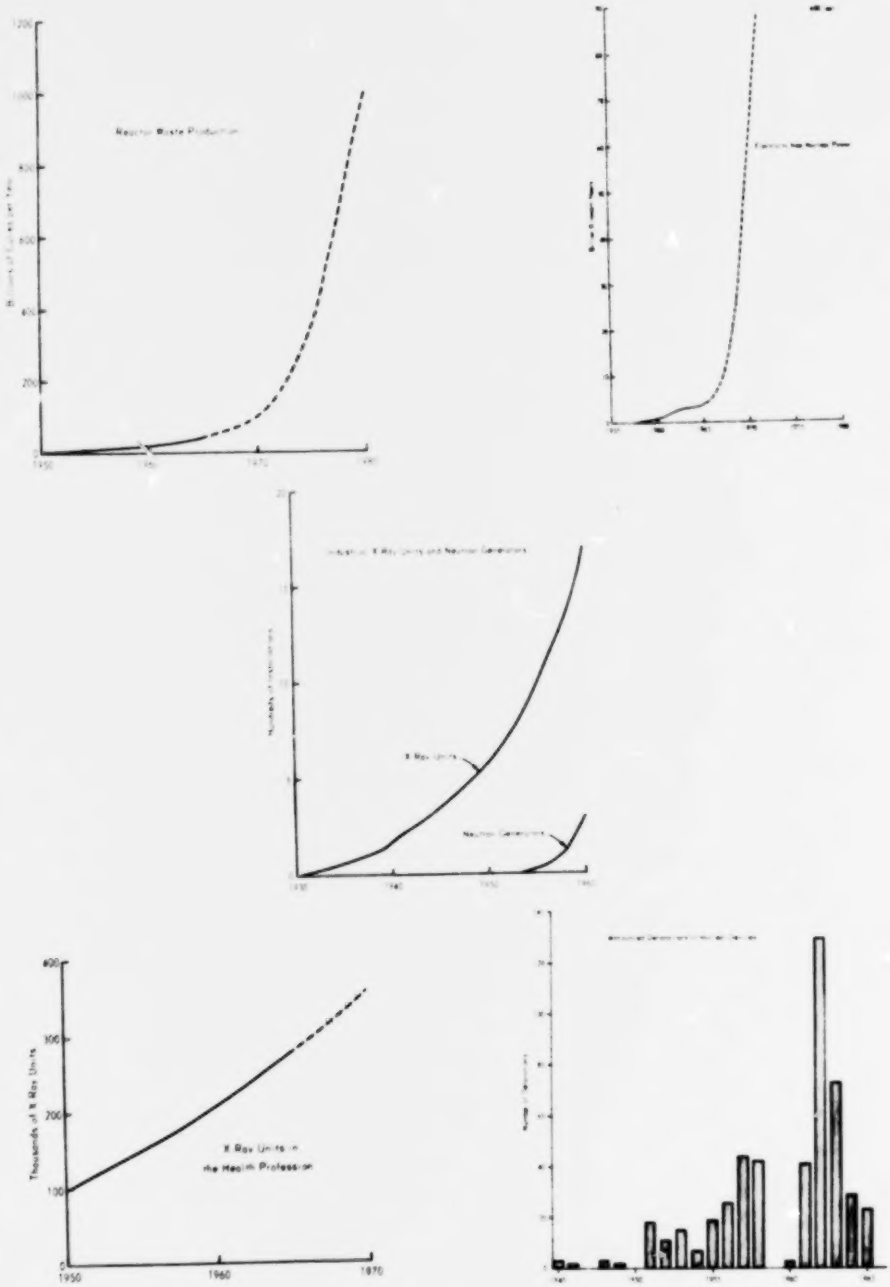

Figure 1.--Increased Use of Radiation

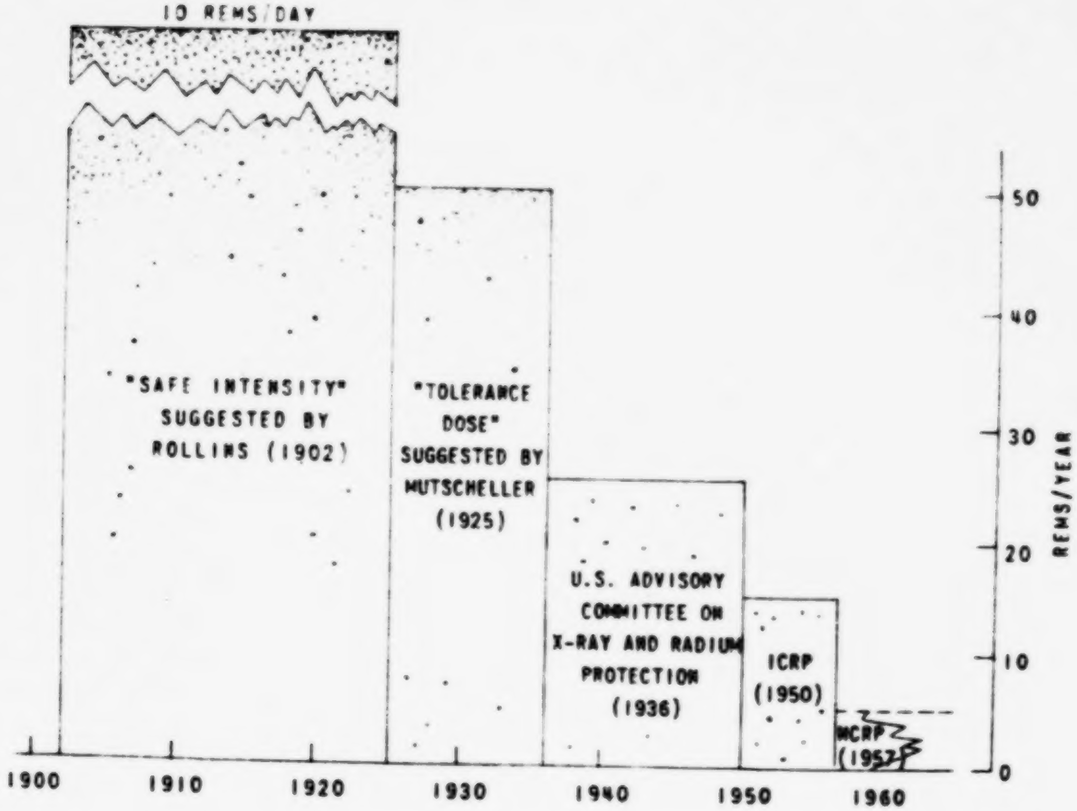

Figure 2.--Decrease in Maximum Permissible Levels
(Occupational Whole-Body Exposure)

TABLE 5--Estimated Annual Genetic Doses (rem/yr) from Man-made Radiation

Medicine and dentistry:		
Diagnostic	0.040-0.240*	
Therapeutic	0.013	
Internal (radionuclides)	0.000	
Total		0.053-0.253
Occupational		0.020
Environmental		0.005
Other		0.002
Total		0.080-0.280†

*The U.S. Public Health Service (X-ray Exposure Study) estimates 0.055 rem/yr from medical and dental x-rays.(4)

† In addition, it is estimated that fallout will contribute from 0.002 to 0.005 rem/yr to the genetically significant per capita dose, depending upon the amount of atmospheric nuclear weapons testing.(2)

The currently recommended Radiation Protection Guide for the genetic dose to general population groups is 5 rems in 30 years, exclusive of natural background and the purposeful exposure of patients. Table 5 shows that exposures associated with medical procedures make a significant contribution to the current dose.

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(11) Personal communication with Cecil Buchanan, Assistant Chief, Isotope Branch, Division of Licensing and Regulation, U.S. Atomic Energy Commission, Washington, D.C. (Mar. 1967).

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(14) The Nuclear Industry (Washington, D.C.: U.S. Atomic Energy Commission, 1966).

(15) Major Activities in the Atomic Energy Program (Washington, D.C.: U.S. Atomic Energy Commission, Jan. 1966).

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Sources of Information in Radiological Health

I. INTRODUCTION

There are many sources of information available to personnel working in the field of radiological health. This outline attempts to list them and give information on their availability. It should be recognized, however, that these are only representative sources. Every publication or source in such a rapidly expanding field cannot possibly be included. Addresses of publishers are at the end of this section.

II. TEXTBOOKS

Textbooks provide valuable basic information in radiological health, though some may deal exclusively with atomic and nuclear physics, others with nomenclature and units, still others with radiation-generating machines and radioactive materials, their use, and the design of facilities for managing them.

The following are some basic reference publications.

A. General Reference and Basic Data:

1. *American Standard, A Glossary of Terms in Nuclear Science and Technology,* (ASA N1.1-1957), published by the American Society of Mechanical Engineers, 1957, $5.00.
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5. Glasstone, Samuel, *Sourcebook on Atomic Energy,* 2nd ed., D. Van Nostrand Co., Inc., 1958, $4.40.
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11. *Radiological Health Handbook,* (revised) Publication No. PB-121784R, Public Health Service, 1960, $3.75.

B. Atomic and Nuclear Physics:

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2. Harvey, Bernard G., *Introduction to Nuclear Physics and Chemistry,* Prentice-Hall, Inc., 1962, $13.25.
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4. Comar, Cyril L., *Radioisotopes in Biology and Agriculture,* McGraw-Hill Book Company, Inc., 1955, $12.00.
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7. Friedlander, Gerhart, and Kennedy, J., *Nuclear and Radiochemistry*, second edition, John Wiley and Sons, Inc., 1964, $10.75.
8. Harvey, Bernard J., *Introduction to Nuclear Physics and Chemistry*, Prentice-Hall, Inc., 1962, $11.25.
9. Korff, Serge A., *Electron and Nuclear Counters*, 2nd ed., D. Van Nostrand Co., Inc., 1955, $1.75.
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3. Eisenbud, Merril, *Environmental Radioactivity*, McGraw-Hill Publishing Co., Inc., 1960, $13.50.
4. Junge, C. E., *Air Chemistry and Radioactivity*, Academic Press, 1963. $13.50.
5. Meneely, George R., and Lind, Shirley M., editors, *Radioactivity in Man*, 2nd ed., Charles C. Thomas, Publisher, 1964. $24.50.
6. Schultz, V., and Klement, A. W., Jr., editors, *Radioecology*, Reinhold Publishing Corp., 1963. $16.50.

J. Radioactive Wastes:

1. Amphlett, C. B., *Treatment and Disposal of Radioactive Wastes*, Pergamon Press, 1961. $12.00.
2. Collins, J. C., *Radioactive Wastes*, John Wiley and Sons, Inc., 1961. $8.95.
3. Glueckauf, E., *Atomic Energy Waste*, (Interscience), John Wiley & Sons, Inc., 1961. $14.00.
4. Saddington, K. and Templeton, W. L., *Disposal of Radioactive Waste*, George Newnes Limited, London, 1958. $3.95.

K. Federal-State Relationships:

1. *Atomic Energy Law Reporter*, Commerce Clearing House, Inc., reports issued weekly, $250.00 per year.

L. Proceedings and Reviews:

1. *Annual Review of Nuclear Science*, Annual Reviews, Inc., $8.50.
2. *Proceedings of the International Conference on the Peaceful Uses of Atomic Energy*, 16 Volumes, United Nations, 1956.
3. *Proceedings of the Second International Conference on the Peaceful Uses of Atomic Energy*, 34 Volumes, United Nations, 1959.
4. *Proceedings of the Third International Conference on the Peaceful Uses of Atomic Energy*, 16 Volumes, United Nations, 1964.

III. CURRENT LITERATURE

Public health workers in radiological health must keep abreast of current literature in a rapidly changing field. The following references represent titles and sources for some of the more recent publications of this type. The same sources may provide still newer information from time to time.

A. Publications of Government Agencies:

1. Handbooks of the National Bureau of Standards:
   Available from the U.S. Government Printing Office, titles and prices of the Handbooks are:
   a. Handbook 48, *Control and Removal of Radioactive Contamination in Laboratories*. 20 cents.
   b. Handbook 49, *Recommendations for Waste Disposal of Phosphorous 32 and Iodine 131 for Medical Users*. 15 cents.
   c. Handbook 51, *Radiological Monitoring Methods and Instruments*. 20 cents.
   d. Handbook 53, *Recommendations for the Disposal of Carbon-14 Wastes*. 15 cents.

e. Handbook 55, *Protection Against Betatron-Synchrotron Radiations up to 100 Million Electron Volts.* 25 cents.
f. Handbook 57, *Photographic Dosimetry of X-rays and Gamma Rays.* 15 cents.
g. Handbook 58, *Radioactive-Waste Disposal in the Ocean.* 20 cents.
h. Handbook 59, *Permissible Dose from External Sources of Ionizing Radiation.* 35 cents.
i. Handbook 63, *Protection Against Neutron Radiation up to 30 Million Electron Volts.* 40 cents.
j. Handbook 64, *Design of Free-Air Ionization Chambers.* 20 cents.
k. Handbook 65, *Safe Handling of Bodies Containing Radioactive Isotopes.* 15 cents.
l. Handbook 66, *Safe Design and Use of Industrial Beta-Ray Sources.* 20 cents.
m. Handbook 69, *Maximum Permissible Body Burdens and Maximum Permissible Concentrations of Radionuclides in Air and Water for Occupational Exposure.* 35 cents.
n. Handbook 72, *Measurement of Neutron Flux and Spectra for Physical and Biological Applications.* 35 cents.
o. Handbook 73, *Protection Against Radiations from Sealed Gamma Sources.* 30 cents.
p. Handbook 75, *Measurement of Absorbed Dose of Neutrons, and of Mixtures of Neutrons and Gamma Rays.* 35 cents.
q. Handbook 76, *Medical X-ray Protection up to 3 Million Volts.* 25 cents.
r. Handbook 77, *Precision Measurement and Calibration,* Volume III, "Optics, Metrology and Radiation." $7.00.
s. Handbook 79, *Stopping Powers for Use with Cavity Chambers.* 35 cents.
t. Handbook 80, *A Manual of Radioactive Procedures.* 50 cents.
u. Handbook 84, *Radiation Quantities and Units* (ICRU Report 10a). 20 cents.
v. Handbook 85, *Physical Aspects of Irradiation* (ICRU Report 10b). 70 cents.
w. Handbook 86, *Radioactivity* (ICRU Report 10c). November 29, 1963. 40 cents.
x. Handbook 87, *Clinical Dosimetry* (ICRU Report 10d). 25 cents.
y. Handbook 88, *Radiobiological Dosimetry* (ICRU Report 10e). 25 cents.
z. Handbook 89, *Methods of Evaluating Radiological Equipment and Materials* (ICRU Report 10f). 35 cents.
aa. Handbook 92, *Safe Handling of Radioactive Materials.* 40 cents.
ab. Handbook 93, *Safety Standards for Non-Medical X-ray and Sealed Gamma-Ray Sources.* 30 cents.
ac. Handbook 97, *Shielding for High-Energy Electron Accelerator Installations.* 30 cents.

2. Reports of the Federal Radiation Council:

The Federal Radiation Council is a statutory body which advises the President on radiation matters directly or indirectly affecting health. FRC helps all Federal agencies formulate radiation standards, establish and execute programs in cooperation with States.

Available from the U.S. Government Printing Office. Titles and prices of Federal Radiation Council Reports are:

a. *Background Material for the Development of Radiation Protection Standards,* Report No. 1, May 1960. 30 cents.
b. *Background Material for the Development of Radiation Protection Standards.* Report No. 2, September 1961. 20 cents.
c. *Health Implications of Fallout from Nuclear Weapons Testing Through 1961.* Report No. 3, May 1962. 15 cents.
d. *Estimates and Evaluation of Fallout in the United States from Nuclear Weapons Testing Conducted Through 1962,* Report No. 4, May 1963. 30 cents.
e. *Background Material for the Development of Radiation Protection Standards,* Report No. 5, July 1964. 20 cents.
f. *Revised Fallout Estimates for 1964-1965 and Verification of the 1963 Predictions,* Report No. 6, October 1964. 25 cents.
g. *Background Material for the Development of Radiation Protection Standards; Protective Action Guides for Strontium-89, Strontium-90, and Cesium-137,* Report No. 7, May 1965. 30 cents.

3. Publications of the Atomic Energy Commission:

a. The Atomic Energy Commission issues a number of very useful reports and periodicals on radiological health. Among these is a publication, *What's Available in the Atomic Energy Literature,* TID-4550, which also lists locations of AEC Depository Libraries.
    (1) *Engineering Materials List: Supplements 1-23,* TID-4100, Division of Technical Information Extension, September 1964.
    (2) *Guide to Atomic Energy Literature for the Civilian Application Program,* TID-4575 (First revision), Division of Technical Information, November 1957.
    (3) *Health Aspects of Nuclear Weapons Testing.* Gordon M. Dunning, available from the U.S. Government Printing Office, 1964. 25 cents.

    (4) *Health and Safety Laboratory Manual of Standard Procedures*, NYO-4700, New York Operations Office, August 1962.
    (5) *Index to Conferences Relating to Nuclear Science*, TID-4043 (revision one), Division of Public Information, May 1964.
    (6) *Nuclear Safety*, available from the U.S. Government Printing Office. $2.50 per year (issued quarterly).
    (7) *Nuclear Science Abstracts*, a publication of the AEC Technical Information Service Extension. Available from the U.S. Government Printing Office. $7.50 per year.

This report, issued twice monthly, contains abstracts of articles pertinent to all areas of nuclear science and technology. In particular, it includes summaries of all nonclassified reports issued by the various AEC installations. Copies of the complete reports are available on loan from AEC Depository Libraries.

b. Reports of the U.S. Atomic Energy Commission to Congress. Available from the U.S. Government Printing Office.
  (1) Sixth Semiannual Report, *Atomic Energy and the Life Sciences*, July 1949. 45 cents.
  (2) Thirteenth Semiannual Report, *Assuring Public Safety in Continental Weapons Tests*, January 1953. 50 cents.
  (3) Twenty-first Semiannual Report, *Radiation Safety and Major Activities in the Atomic Energy Programs*. January 1957. $1.25.
  (4) Twenty-fifth Semiannual Report, *Atomic Industrial Progress and Second World Conference*. January 1959. $1.25.
  (5) Annual Reports, *Major Activities in the Atomic Energy Programs:*
    (a) January 1960 (for 1959). $2.00
    (b) January 1961 (for 1960). $1.75
    (c) January 1962 (for 1961). $2.00
    (d) January 1963 (for 1962). $1.75
    (e) January 1964 (for 1963). $2.00
    (f) January 1965 (for 1964). $1.75

4. Environmental Health Series; Radiological Health: Single free copies may be obtained from the Public Health Service.
  a. No. 999-RH-1, *Factors Influencing Strontium-90 in Milk from the Brainerd, Minnesota Milkshed*. 1962.
  b. No. 999-RH-2, *Rapid Methods for Estimating Fission Product Concentration in Milk*. 1963.
  c. No. 999-RH-3, *Studies of the Fate of Certain Radionuclides in Esturine and Other Aquatic Environments*. 1963.
  d. No. 999-RH-4, *Mathematical Programming Models for Selection of Diets to Minimize Weighted Radionuclide Intake*. 1963.
  e. No. 999-RH-5, *Radionuclide Analysis of Gamma-Ray Spectra by Stepwise Multiple Regression*. 1963.
  f. No. 999-RH-6, *Farming Practices and Concentrations of Fission Products in Milk*. 1964.
  g. No. 999-RH-7, *An Environmental Surveillance Laboratory for Radionuclide Analysis*. 1964.
  h. No. 999-RH-8, *X-ray Equipment Survey in Polk County, Florida*. 1964.
  i. No. 999-RH-9, *An Emanation System for Determining Small Quantities of Radium-226*. 1964.
  j. No. 999-RH-10, *Procedures for Determination of Stable Elements and Radionuclides in Environmental Samples*. 1965.
  k. No. 999-RH-11, *Radiochemical Determination of Uranium in Environmental Media by Electrodeposition*. 1965.
  l. No. 999-RH-12, *Radioactive Decay Correction Factors*. 1965.
  m. No. 999-RH-13, *Behavior of Certain Radionuclides Released into Fresh Water Environments*. 1965.
  n. No. 999-RH-14, *Iodine-131 in Children's Thyroids From Environmental Exposure*. 1965.
  o. No. 999-RH-15, *Quality Control of Radioactivity - Counting Systems*. 1965.
  p. No. 999-RH-16, *Medical Uses of Radium and Radium Substitutes*.
  q. No. 999-RH-17, *Radionuclide Analysis of Large Numbers of Food and Water Samples*.
  r. No. 999-RH-18, *Mortality of New England Dentists, 1921-1960*.
  s. No. 999-RH-19, *Soil and Sediment Analysis; P[illegible]ration of Samples for Environmental [illegible]diation Surveillance*.
  t. No. 999-RH-20, *Standards of Performance for Film Badge Services*.
  u. No. 999-RH-21, *A Computer Program for the Analysis for Gamma Ray Spectra by the Method of Least Squares*.
  v. No. 999-RH-22, *State Legislation and Regulations Involving Ionizing Radiation*. 1965.

5. Handbooks and Special Reports.
   a. *Analyses of Environmental Samples: Chemical and Radiochemical Procedures.* Free copies available from the Northeastern Radiological Health Laboratory.
   b. *Analytical Methods of the Southwestern Radiological Health Laboratory.* Free copies available from the Southwestern Radiological Health Laboratory.
   c. *Calculation of Distance Factors for Power and Test Reactor Sites.* TID-14844, March 1962. $1.00. U.S. Department of Commerce.
   d. Medical Research Council. *The Hazards to Man of Nuclear and Allied Radiations* (a second report), December 1960. 7 shillings. Available from Her Majesty's Stationery Office.
   e. *Nuclear Reactors and Earthquakes.* TID-7024, August 1963. $5.00. Available from U.S. Department of Commerce.
   f. *Oak Ridge National Laboratory, Master Analytical Manual.* TID-7015 and supplements 1 through 6, 1958. $9.00. Available from the Division of Technical Information.
   g. *Procedures for the Determination of Stable Elements and Radionuclides in Environmental Samples.* Free copies available from Southeastern Radiological Health Laboratory.
   h. *Radionuclide Analysis of Environmental Samples—A Laboratory Manual of Methodology.* Technical Report R59-6. Free copies available from Robert A. Taft Sanitary Engineering Center.
   i. *Report of the United Nations Scientific Committee on the Effects of Atomic Radiation.* Supplement No. 16 (A/5216), 1962. $5.00. Available from United Nations.
   j. *Siting of Nuclear Power Reactors.* 1959. $2.00. Available from Atomic Industrial Forum, Inc.
   k. Smyth, H. D., *A General Account of the Development of Methods of Using Atomic Energy for Military Purposes Under the Auspices of the United States Government* (1940-1945), 1945. 40 cents. Available from the U.S. Government Printing Office.
   l. Straub, C. P., *Low-Level Radioactive Wastes; Their Handling, Treatment, and Disposal.* $1.50. Available from the U.S. Government Printing Office.

B. Special Reports

1. National Academy of Sciences-National Research Council Publications:
   a. Available from National Academy of Sciences-National Research Council:
      (1) *Behavior of Radioactive Fallout in Soils and Plants.* Publication 1092, 1963.
      (2) *Biological Effects of Atomic Radiation. Summary Reports.* 1960. Free.
      (3) *Disposal of Low-Level Radioactive Waste into Pacific Coastal Waters.* Publication 985, 1962. $2.00.
      (4) *Effects of Inhaled Radioactive Particles.* Publication 848, 1961. Free.
      (5) *Effects of Ionizing Radiation on the Human Hematopoietic System.* Publication 875, 1961. Free.
      (6) *Internal Emitters.* Publication 883, 1961. Free.
      (7) *Long-term Effects of Ionizing Radiations from External Sources.* Publication 849, 1961. Free.
      (8) *Measurements and Standards of Radioactivity.* Publication 573. $1.75.
      (9) *Multichannel Pulse Height Analyzers.* Publication 467, 1957.
      (10) *Radioactive Contamination of Materials Used in Scientific Research.* Publication 895. $2.00.
      (11) *Radioactive Waste Disposal from Nuclear-Powered Ships.* Publication 658, 1959.
      (12) *Radioactive Waste Disposal into Atlantic and Gulf Coastal Waters.* Publication 655, 1959.
      (13) *Source Material for Radiochemistry (Rev. 1).* Publication 825. Free.
      (14) *Status Report on Standardization of Radionuclides in the United States.* Publication 570. Free.
      (15) *Treatment of Radiation Injury.* The Publication 1134, 1963. $1.00.
   b. Available from the U.S. Department of Commerce:
      (1) *Radiochemistry of the Elements.*
   c. Available from the U.S. Government Printing Office:
      (1) *Nuclear Data Tables.* Parts 1-4 (1959-1960). $1.00.
      (2) *Nuclear Data Sheets.* Volumes 1-5 (1958-1963). $18.00 per volume—paper stock. $24.00 per volume—card stock.

2. Recommendations of the International Commission on Radiological Protection. Available from the Macmillan Co.
   - a. *Permissible Dose for Internal Radiation,* (ICRP Publication 2), 1959. Contains reprint of ICRP Publication 1.
   - b. *Protection Against X-rays up to Energies of 3 MeV and Beta- and Gamma-Rays from Sealed Sources,* (ICRP Publication 3), 1960. Contains reprint of ICRP Publication 1.
   - c. *Protection Against Electromagnetic Radiation above 3 MeV and Electrons, Neutrons and Protons.* (ICRP Publication 4), 1964.
   - d. *Handling and Disposal of Radioactive Materials in Hospitals and Small Establishments.* (ICRP Publication 5), 1964.
   - e. *Radiation Protection.* As Amended 1959 and Revised 1962, (ICRP Publication 6), 1964.
3. Publications of the International Atomic Energy Agency: Available from National Agency for International Publications, Inc.
   - a. *Publications in the Nuclear Sciences.* (A complete catalog published yearly.)
   - b. *Radioactive Waste Disposal into the Sea.* Safety Series No. 5, September, 1961.
   - c. *Regulations for the Safe Transport of Radioactive Materials.* Safety Series No. 6. $1.50.
   - d. *Safe Handling of Radioisotopes.* Safety Series No. 1, $1.50.
4. Other Reports:
   - a. *Exposure to Radiation in an Emergency.* (NCRP Report 29), The University of Chicago, January, 1962. 50 cents.
   - b. Reports of The American College of Radiology.
   - c. Reports of The American Hospital Association.
   - d. Reports of The American Public Health Association, Inc.
   - e. Reports of The National Fire Protection Association.
   - f. Reports of the United Nations.

C. Periodicals

1. American Journal of Roentgenology, Radium Therapy, and Nuclear Medicine, published monthly by Charles C. Thomas, Publisher, $15.00 per year.
2. Analytical Chemistry, published monthly by the American Chemical Society. $5.00 per year.
3. Bulletin of the Atomic Scientists, published monthly by the University of Chicago Press. $6.00 per year.
4. Health Physics, Pergamon Press, Inc. $60.00 per year.
5. Isotopes and Radiation Technology. Available from the U.S. Government Printing Office. $2.00 per year (issued quarterly).
6. Journal of Cellular and Comparative Physiology, published bimonthly by Westar Institute of Anatomy and Biology.
7. Nuclear Instruments and Methods, published monthly by North-Holland Publishing Co. $17.00 a volume.
8. Nucleonics, published monthly by the McGraw-Hill Publishing Co., Inc. $8.00 per year.
9. Public Health Engineering Abstracts. Available from the U.S. Government Printing Office. $2.00 per year (published monthly).
10. Radiation Research, published monthly by Academic Press, Inc. $14.00 per volume.
11. Radiologic Technology, published bimonthly by Williams and Wilkins Co. $3.00 per year.
12. Radiological Health Data. For sale by the U.S. Government Printing Office. $6.50 per year (published monthly).
13. Radiology, published monthly by the Radiological Society of North America. $10.00 per year.
14. Reviews of Modern Physics, published quarterly by the American Institute of Physics, Inc. $6.00 per year.
15. Science, published weekly by the American Association for Advancement of Science. $8.50 per year.

IV. FILMS

Many films covering one or more of the several phases of radiological health have been produced. These are an excellent source of information, especially valuable as teaching aids. Following are several film catalogs describing current films. Addresses of the companies and agencies are listed in the Index of Mailing Addresses.

A. The Atomic Energy Commission maintains ten motion picture film libraries in the United States. Film catalogs are available from the Division of Public Information.

B. *Filmstrip Guide.* H. W. Wilson Co. $14.00.

C. *Educational Film Guide.* H. W. Wilson Co. $20.00.

D. *Public Health Service Film Catalog.* PHS Publication 776, U.S. Department of Health, Education and Welfare. Available from the U.S. Government Printing Office. 75 cents.

E. *Film Reference Guide for Medicine and Allied Sciences*, PHS Publication 487, U.S. Department of Health, Education and Welfare. Available from the U.S. Government Printing Office. $2.50.

F. Some films (listed in IV. D and E) used in training courses conducted by the Training and Manpower Development Program of the National Center for Radiological Health are:

1. "A is for Atom"
2. "Basic Principles of Power Reactors"
3. "Fundamentals of Radioactivity," Parts I and II
4. "Nuclear Radiation Uses in Industry"
5. "Nuclear Radiation Uses in Medicine"
6. "Radiation and Public Health"
7. "Radiation: Physician and Patient"
8. "Radiation Protection in Nuclear Medicine"
9. "Radium Decontamination"

V. TRAINING PROGRAMS

Many colleges and universities now offer courses in radiological health. These vary both in the depth of the material covered and the length of the courses. Here are sources of information on curricula planning and the scheduling of training courses.

1. *Atomic Energy Commission Special Fellowships in Health Physics*, AEC, Oak Ridge Institute of Nuclear Studies.
2. *Bulletin of Courses in Environmental Health Sciences and Engineering Training Program*, Public Health Service, Robert A. Taft Sanitary Engineering Center. Issued annually.
3. *Education and Nuclear Energy*, STI/PUB/19, International Atomic Energy Agency.
4. *Post-Graduate Training in the Public Health Aspects of Nuclear Energy*, Technical Report Series No. 154, World Health Organization. 1958.
5. *Schedule of OCD Training Courses*, Information Bulletin, Office of Civil Defense, U.S. Department of the Army.
6. *Syllabus of Courses*, the Isotope School, Atomic Energy Research Establishment.
7. *Training Course Announcement*, Division of State and Licensee Relations, Atomic Energy Commission.
8. *Training Course Schedule*, AEC, Special Training Division.
9. *Training in Radiological Health and Safety*, Her Majesty's Stationery Office. 1960.
10. *Training in Radiological Protection: Curricula and Programming*, Technical Reports Series No. 31, International Atomic Energy Agency. 1964.
11. *University Curriculums and Fellowships in Radiological Health*, Public Health Service, Division of Radiological Health. March 1964.

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U.S. Atomic Energy Commission
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Public Health Service

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GPO 917-216

Atomic Structure

I. INTRODUCTION

To understand the various processes of radioactive decay, it is necessary to understand the structure of the atom.

This review presents the fundamental concepts of stable atoms, the symbols, units, and terminology commonly used in nuclear physics and radiological health and the concepts of ions, isotopes, and nuclear binding energy.

II. STRUCTURE OF THE ATOM

A. General

The atom is the simplest unit into which an element can be divided and still retain the properties of the original element. Atoms of all elements are made of three primary subatomic components: electrons, protons, and neutrons. These three primary components are so arranged that the atom may be thought of as consisting of two main parts, the nucleus and the electron cloud. A graphical representation of several atoms is shown in Figure 1.

1. THE NUCLEUS

The nucleus is the central portion of the atom and contains two of the three components: the protons and neutrons. They make up practically all of the mass of the atom. Protons and neutrons are often referred to as nucleons. Thus, we consider the nucleus to consist of various combinations of protons and neutrons. It is easier to understand the following discussion if these combinations are thought of as being held together by some force sufficiently great to overcome the coulomb force of repulsion that exists between the positively charged protons. Table 1 lists the subatomic particles as to whether they are fundamental components, generated particles, or derived particles.

2. THE ELECTRON CLOUD

The electron cloud surrounds the nuclear portion of the atom and contains the electrons which are in motion about the nucleus. Since the atom as a whole is electrically neutral, the number of electrons in the electron cloud is equal to the number of positive charges (protons) in the nucleus.

The results of various studies[1] of the electron arrangements in atoms indicate that electrons in an atom are grouped into shells each having a definite energy and a definite maximum number of electrons. It is customary to visualize electrons of the atom as moving about the nucleus in orbits much as the planets revolve in elliptical paths about the sun. Although this mental image is not entirely accurate, it provides a useful model for understanding the atom. These well-defined orbits (or shells) are often referred to as energy (quantum) levels since a definite minimum amount of energy is required to remove an electron from a particular shell. Each shell is denoted by a number in order of decreasing binding energy or, more commonly, by a letter

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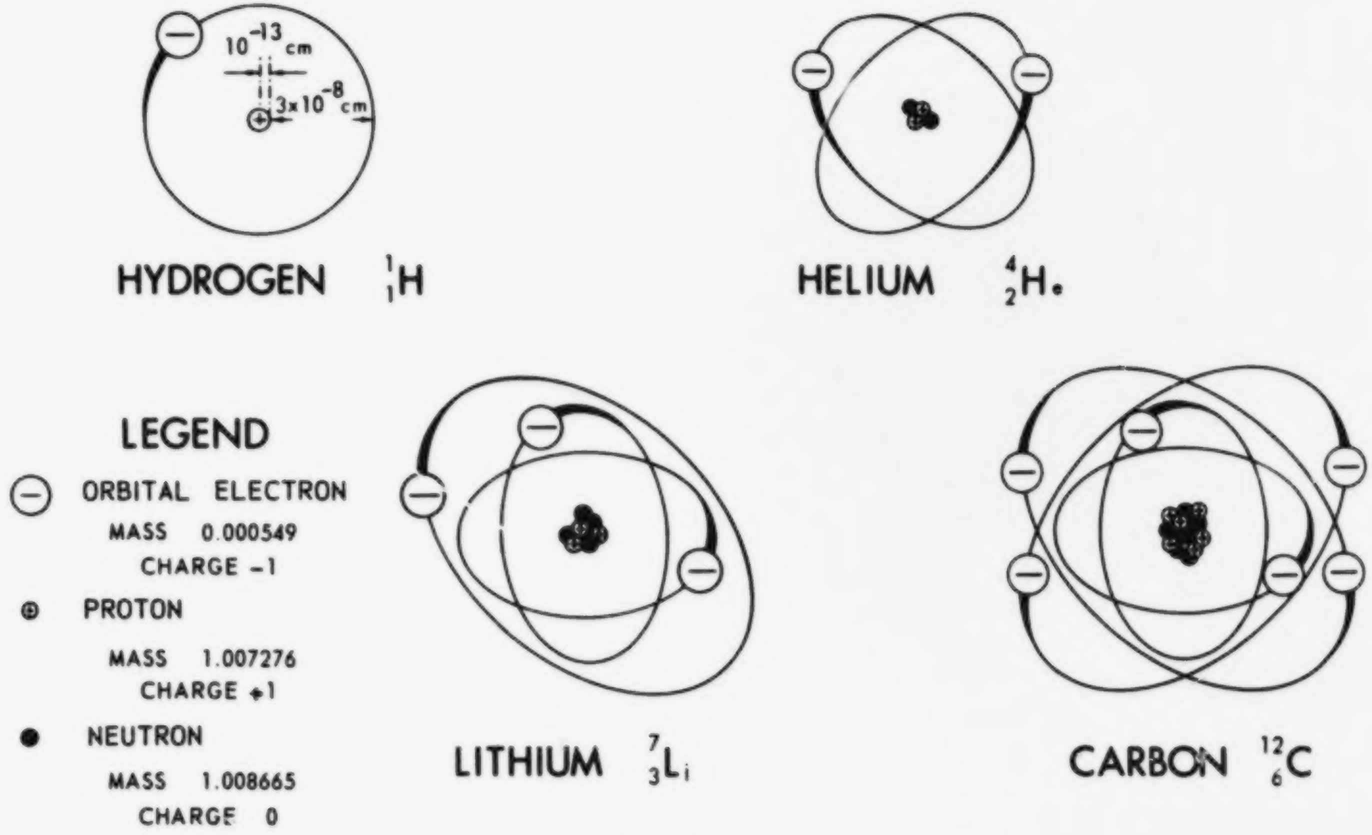
[1] Our knowledge of the electric configurations of atoms has come chiefly from studies of atomic spectra and variation in chemical and physical properties with atomic number.

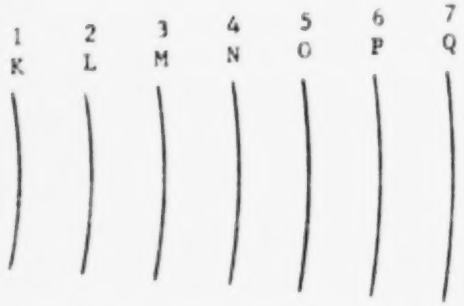

10^{-13} cm
3×10^{-8} cm
HYDROGEN $^{1}_{1}H$
HELIUM $^{4}_{2}He$
LEGEND
ORBITAL ELECTRON
MASS 0.000549
CHARGE -1
PROTON
MASS 1.007276
CHARGE +1
NEUTRON
MASS 1.008665
CHARGE 0
LITHIUM $^{7}_{3}Li$
CARBON $^{12}_{6}C$

Atomic Structure

Table 1 Atomic Components

Unit	Symbol (Common & Alternate)	Relative Mass in Atomic Mass Units	Charge (Electrical)		
			coulombs	esu	relative
Fundamental					
Proton	p, $^{1}_{1}H$	1.007276	1.6×10^{-19}	4.8×10^{-10}	+
Electron	e, β^{-}	0.000549	1.6×10^{-19}	4.8×10^{-10}	-
Neutron	n, $^{1}_{0}n$	1.008665	0	0	0
Generated					
Positron	e^{+}, β^{+}	0.000549	1.6×10^{-19}	4.8×10^{-10}	+
Derived					
Deuteron	d, $^{2}_{1}H$	2.013553	1.6×10^{-19}	4.8×10^{-10}	+
Alpha	α, $^{4}_{2}He$	4.001502	3.2×10^{-19}	9.6×10^{-10}	2+

designation from K to Q, as shown below.



The first shell is designated as K, the second as L, the third M, and so on through the alphabet to Q, or seventh shell, which is the outermost shell ever found in any of the presently known elements. Table 2 lists the elements and the various number of electrons in each shell.

B. Ions and Ionization

All atoms in their natural, lowest energy state have the same number of electrons as protons. Consequently, they carry no residual electric charge. When atoms combine to form compounds, the resulting molecule has the same total number of electrons as it does protons; and the molecule is also electrically neutral. Since the electron is the lightest part of the atom, and since each electron is not bound as tightly to the nucleus as the protons and neutrons are bound to each other, it is more mobile and can be removed from the atom or molecule without expending much energy. As soon as an electron is removed from an atom or molecule, the resulting component has an electric charge because the protons are then in excess of the electrons. The residual atom, remaining after the removal of an electron, and the electron

Table 2. The Atomic Structures

Period	Atom	Atomic No.	K Shell 1	L Shell 2	M Shell 3	N Shell 4	O Shell 5	P Shell 6	Q Shell 7
1	H	1	1						
	He	2	2						
2	Li	3	2	1					
	Be	4	2	2					
	B	5	2	3					
	C	6	2	4					
	N	7	2	5					
	O	8	2	6					
	F	9	2	7					
3	Ne	10	2	8					
	Na	11	2	8	1				
	Mg	12	2	8	2				
	Al	13	2	8	3				
	Si	14	2	8	4				
	P	15	2	8	5				
	S	16	2	8	6				
	Cl	17	2	8	7				
4	A	18	2	8	8				
	K	19	2	8	8	1			
	Ca	20	2	8	8	2			
	Sc	21	2	8	9	2			
	Ti	22	2	8	10	2			
	V	23	2	8	11	2			
	Cr	24	2	8	13	1			
	Mn	25	2	8	13	2			
	Fe	26	2	8	14	2			
	Co	27	2	8	15	2			
	Ni	28	2	8	16	2			
	Cu	29	2	8	18	1			
	Zn	30	2	8	18	2			
	Ga	31	2	8	18	3			
	Ge	32	2	8	18	4			
	As	33	2	8	18	5			
	Se	34	2	8	18	6			
	Br	35	2	8	18	7			
5	Kr	36	2	8	18	8			
	Rb	37	2	8	18	8	1		
	Sr	38	2	8	18	8	2		
	Y	39	2	8	18	9	2		
	Zr	40	2	8	18	10	2		
	Nb	41	2	8	18	12	1		
	Mo	42	2	8	18	13	1		
	Tc	43	2	8	18	14	1		
	Ru	44	2	8	18	15	1		
	Rh	45	2	8	18	16	1		
	Pd	46	2	8	18	18	0		
	Ag	47	2	8	18	18	1		
	Cd	48	2	8	18	18	2		
	In	49	2	8	18	18	3		
	Sn	50	2	8	18	18	4		
	Sb	51	2	8	18	18	5		
	Te	52	2	8	18	18	6		
	I	53	2	8	18	18	7		
6	Xe	54	2	8	18	18	8		
	Cs	55	2	8	18	18	8	1	
	Ba	56	2	8	18	18	8	2	
	La	57	2	8	18	18	9	2	
Rare Earths	Ce	58	2	8	18	19	9	2	
	Pr	59	2	8	18	20	9	2	
	Nd	60	2	8	18	22	8	2	
	Pm	61	2	8	18	23	8	2	
	Sm	62	2	8	18	24	8	2	
	Eu	63	2	8	18	25	8	2	
	Gd	64	2	8	18	25	9	2	
	Tb	65	2	8	18	26	9	2	
	Dy	66	2	8	18	28	8	2	
	Ho	67	2	8	18	29	8	2	
	Er	68	2	8	18	30	8	2	
	Tm	69	2	8	18	31	8	2	
	Yb	70	2	8	18	32	8	2	
	Lu	71	2	8	18	32	9	2	
	Hf	72	2	8	18	32	10	2	
	Ta	73	2	8	18	32	11	2	
	W	74	2	8	18	32	12	2	
	Re	75	2	8	18	32	13	2	
	Os	76	2	8	18	32	14	2	
	Ir	77	2	8	18	32	15	2	
	Pt	78	2	8	18	32	17	1	
	Au	79	2	8	18	32	18	1	
	Hg	80	2	8	18	32	18	2	
	Tl	81	2	8	18	32	18	3	
	Pb	82	2	8	18	32	18	4	
	Bi	83	2	8	18	32	18	5	
	Po	84	2	8	18	32	18	6	
	At	85	2	8	18	32	18	7	
7	Rn	86	2	8	18	32	18	8	
	Fr	87	2	8	18	32	18	8	1
	Ra	88	2	8	18	32	18	8	2
	Ac	89	2	8	18	32	18	9	2
Rare Earths	Th	90	2	8	18	32	19	9	2
	Pa	91	2	8	18	32	20	9	2
	U	92	2	8	18	32	21	9	2
	Np	93	2	8	18	32	22	9	2
	Pu	94	2	8	18	32	23	9	2
	Am	95	2	8	18	32	25	8	2
	Cm	96	2	8	18	32	25	9	2
	Bk	97	2	8	18	32	26	8	2
	Cf	98	2	8	18	32	27	9	2
	Es	99	2	8	18	32	28	9	2
	Fm	100	2	8	18	32	29	9	2
	Md	101	2	8	18	32	30	9	2
	No	102	2	8	18	32	31	9	2
	Lw	103	2	8	18	32	32	9	2

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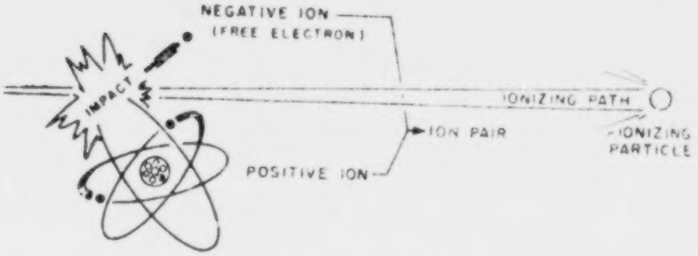

Figure 2. Ionization

itself are called ions. These are shown in Figure 2.

Ions may be charged positively or negatively and may exist in crystals, liquids, or in gases. Positive ions are produced by removing electrons from neutral atoms and molecules, whereas negative ions are created when electrons attach themselves to neutral atoms or molecules. Negative ions are also free electrons and electrons that have been removed from atoms and molecules. The magnitude of the electrical charge on the ion, in terms of electron units, is equal to the number of electrons removed from or added to a neutral atom or molecule. The magnitude of electrical charge on an ion may be calculated in terms of coulombs or electrostatic units by multiplying the number of electrons exchanged (gained or lost) by the charge on each electron in coulombs or electrostatic units. This value is given in the table of atomic components.

As shown in Table 1, the relative charge on an ion in terms of electron units is called the valence of the ion. Recently it has become common practice to consider a free electron as an ion. Ions are, therefore, defined as free electrons or atoms or groups of atoms carrying an electric charge.

Ionization is the process of producing ions. Anything which is able to bring about the removal of electrons from neutral atoms or molecules is capable of causing ionization or an ionizing event. Consequently, the ionizing event is spoken of as producing an ion pair which consists of the removed electron and the residual positive ion.

III. ATOMIC NOMENCLATURE

Before proceeding, it will greatly simplify the discussion of the atom to introduce a consistent set of definitions and symbols.

A general expression for representing a particular element is $^{A}_{Z}X$.

A. Chemical Symbols

In the general expression above, the X represents the chemical symbol of the element and is used for brevity and convenience. In referring to a specific element, the chemical symbol usually consists of the initial letter[2] (capitalized) in the name of the element or the initial letter followed by an appropriate second letter. For example, carbon is represented by C, chlorine by Cl, calcium by Ca, Oxygen by O, and hydrogen by H.

B. Atomic Number (Z)

The Z refers to the number of protons present in the atomic nucleus. This number is always the same as the number of positive charges and is called the atomic number of the element. Since atoms are electrically neutral, the number of electrons in the electron cloud is the same as the number of protons within the nucleus. Each element has an atomic number which is characteristic of the element and is synonymous with the chemical symbol. There are currently 103 elements, 103 symbols, and 103 atomic numbers. Consequently, an element may be designated in any one of the three ways by name, by number, or by letter symbol.

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[2] When this system was adopted (1779-1898), the Latin names for several metals were in common use. This is reflected in such symbols as Fe (from the Latin ferrum) for iron, Cu (cuprum) for copper, Au (aurum) for gold, Ag (argentum) for silver, Sn (stannum) for tin, and Sb (stibium) for antimony.

C. Mass Number (A)

The A in the general expression ($^{A}_{Z}X$) represents the total number of particles (protons and neutrons) in the nucleus and is referred to as the mass number. The mass number is approximately equal to the atomic weight of an element.

D. Neutron Number (N)

The number of neutrons in the nucleus is called the neutron number (N) and is equal to the mass number minus the atomic number (A-Z).

1. ISOTOPES

When the atoms of an element have the same number of protons but a different number of neutrons, they are said to be isotopes of the element. Such atoms are, in general, indistinguishable chemically. The chemical properties of the isotopes of an element are the same because the chemical properties depend on the number and arrangement of (orbital) electrons surrounding the nucleus which, in turn, is determined by the number of protons in the nucleus.

For example, there are three naturally-occurring isotopes of oxygen, namely, $^{16}_{8}O$, $^{17}_{8}O$, and $^{18}_{8}O$. The isotope which has a mass number of 16 occurs most abundantly. Another example is hydrogen which has three isotopes, $^{1}_{1}H$, $^{2}_{1}H$, and $^{3}_{1}H$. This element is so significant that its three isotopes are sometimes called by distinct names. The isotope of mass number 1 is called hydrogen, that of 2 is deuterium, and that of 3 is tritium.

IV. SIZE OF ATOMIC AND SUBATOMIC PARTICLES

A. Radius of the Nucleus

The radius of the nucleus of various nuclides, determined by different experimental methods, are generally in good agreement with each other. Except for the elements of lowest mass number, the size of the nucleus can be expressed by the following empirical equation:

$$r = r_o A^{1/3}, \text{ where}$$

r = the radius of the nucleus of an isotope

r_o = a constant of about 1.37×10^{-13} cm

A = the mass number.

Figure 3 shows the extent to which the equation holds for specific elements.

1. NUCLEAR DENSITY

Since the radius is proportional to the cube root of the mass number and the volume of a sphere (volume = $4/3\,\pi\, r^3$) is proportional to the cube of the radius, the volume of the nucleus is, consequently, directly proportional to the mass number and, hence, to its actual mass. Thus, since all atomic nuclei are composed of the same constituents, namely, neutrons and protons, they have essentially the same density. This density can easily be determined since the protons (and neutrons) have a mass of about 1.67×10^{-24} gm. The radius (A = 1) from the equation $r = r_0 A^{1/3}$ is about 1.37×10^{-13} cm. The density is, thus, approximately 10^5 tons/mm^3.

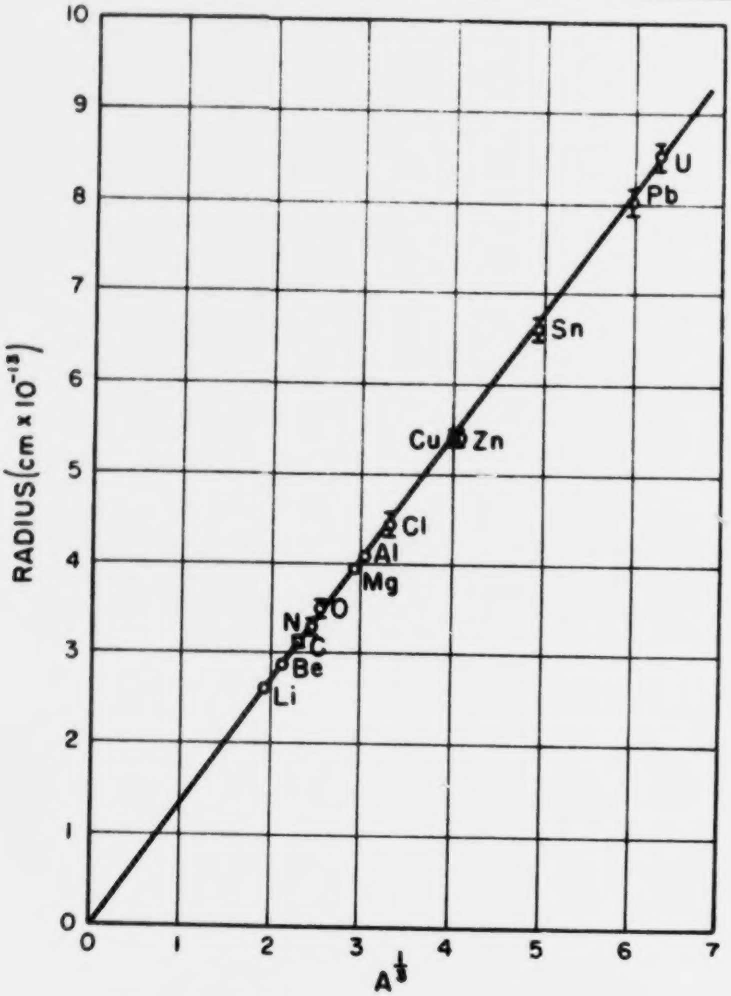

Figure 3. Radius of the Atom and its Nucleus

B. Radius of the Atom

The radius of the entire atom is much larger than the radius of the nucleus, as shown in Table 3.

Table 3. Radius of the Atom and its Nucleus

Element	Radius of Atom	Radius of Nucleus
Carbon	.77 × 10^{-8} cm	3.8 × 10^{-13} cm
Aluminum	1.45 × 10^{-8} cm	4.6 × 10^{-13} cm
Argon	1.5 × 10^{-8} cm	4.1 × 10^{-13} cm
Tin	1.4 × 10^{-8} cm	7.4 × 10^{-13} cm
Lead	1.6 × 10^{-8} cm	7.8 × 10^{-13} cm

For example, the radius of the argon atom is about 1.5×10^{-8} cm, whereas the radius of the nucleus is only 4.1×10^{-13} cm. Thus, the radius of the argon atom is about 40,000 times larger than the radius of its nucleus. This difference between the size of the atom and its nucleus indicates that a great deal of the atom is empty space. Considering the size of the atom at about 10^{-8} and the nucleus at 10^{-13}, calculations show that only about one-trillionth (10^{-12}) of the volume of an atom is occupied by material particles.

V. UNITS OF NUCLEAR MASS AND ENERGY

A. Atomic Mass Unit (amu)

To appreciate the meaning of the atomic mass unit, consider the relative mass of the nuclear particles as shown in Table 4.

Since the mass of an individual nuclear particle is very small and, when expressed in grams, involves the use of unwieldly negative exponents, the system of atomic mass units (amu) was developed using the most plentiful isotope of oxygen ($^{16}_{8}O$) as a reference atom. The present standard utilized carbon-12 as the reference atom. The arbitrary value (weight) of 12 units has been assigned to carbon-12 ($^{12}_{6}C$). The mass of all other atoms is based on a unit which is 1/12 the weight of the carbon-12 atom. The mass of the lightest isotope of hydrogen is approximately 1 (1.007825 amu).

The weight in grams of an isotope numerically equal to its atomic mass is called a gram-atomic weight. Since gram-atomic masses are relative magnitudes, proportional to the actual masses of the individual atoms, it follows that one gram-atomic weight of any substance contains a definite number of atoms. The number of atoms in one gram-atomic mass is known as Avogadro's number (N_o) and is equal to 6.023×10^{23} atoms/gram-atomic weight.

The mass of any atom in grams can thus be found by dividing the gram-atomic mass of the isotope by Avogadro's number. For example, the mass of an atom of carbon-12 is its gram-atomic mass (12 gm.) divided by Avogadro's number (N_o).

$$\text{Mass of } ^{12}C = \frac{12 \text{ gm}}{6.023 \times 10^{23}}$$

$$= 1.99 \times 10^{-23} \text{gm}/^{12}C \text{ atom}$$

Since ^{12}C is equal to 12 atomic mass units (amu), the mass of 1 amu would be one-twelfth of the mass of a carbon-12 atom.

$$1 \text{ amu} = \frac{1.99 \times 10^{-23} \text{ gm}/^{12}C}{12 \text{ amu}/^{12}C}$$

$$= 1.66 \times 10^{-24} \text{gm}$$

Table 4. Relative Mass of Nuclear Particles

Particle	Weight in Grams	Number in One Gram	Symbol	Charge
Neutron	1.67482×10^{-24}	5.9724×10^{23}	n	0
Proton	1.67252×10^{-24}	5.9790×10^{23}	p	+1
Electron	9.1091×10^{-28}	1.0978×10^{27}	e	-1

B. The Erg

To facilitate an understanding of the enormous energy contained in a small mass, consider the energy produced if one amu (1.66×10^{-24} gm) is converted into energy. This conversion can be expressed by Einstein's famous equation relating mass and energy.

$$E = mc^2$$

where: m = the mass in grams

c = the velocity of light

$$E = (1.66 \times 10^{-24} \text{gm})(3 \times 10^{10} \text{cm/sec})^2$$

$$= 1.494 \times 10^{-3} \text{gm-cm}^2/\text{sec}^2$$

The unit $\text{gm-cm}^2/\text{sec}^2$ is frequently encountered in physics and radiological health work. For brevity, it is commonly referred to as an erg, thus:

$$1.494 \times 10^{-3} \text{gm-cm}^2/\text{sec}^2 = 1.49 \times 10^{-3} \text{ ergs}$$

C. The Electron Volt

Another convenient unit of energy is the electron volt. An electron volt is defined as the amount of energy acquired by an electron when it is accelerated in an electric field which is produced by a potential difference of one volt. Since the work done by a difference of potential, V, acting on a charge, e, is eV, then, because $e = 1.6 \times 10^{-19}$ coulomb, it is possible to calculate as follows:

$$1 \text{ eV} = 1.6 \times 10^{-19} \text{coulomb} \times 1 \text{ volt}$$

$$= 1.6 \times 10^{-19} \text{joule}$$

$$1 \text{ eV} = 1.6 \times 10^{-12} \text{ergs}$$

Since the electron volt is a very small amount of energy, it is more commonly expressed in thousands of electron volts (keV) or millions of electron volts (MeV).

1 million electron volts (MeV)

= 1,000,000 electron volts

1 kilo-electron volts (keV)

= 1000 electron volts

It is now possible to express the atomic mass unit in MeV. Since 1 amu = 1.49×10^{-3} ergs, and 1 MeV = 1.60×10^{-6} ergs, 1 amu = 931 MeV.

Therefore, if one amu could be completely transformed into energy, 931 MeV would result. To gain a concept of the magnitude of the electron volt, it may be pointed out that one MeV is the energy necessary to lift a milligram weight one-millionth of a centimeter.

VI. NUCLEAR BINDING ENERGY

The atomic mass of a nuclide can be understood in terms of the masses of its constituent particles and a quantity called the nuclear binding energy. It might be thought that the mass of an atom would be equal to sum of the masses of the individual particles composing the atom. However, precise measurements of atomic masses show that the mass of an atom is always less than the sum of its constituent particles (as measured in the free state). This difference is called the mass defect and is the amount of mass which would be converted into energy and released (to hold the nucleons together) if an atom were assembled from free neutrons, protons, and electrons. Since this is also the amount of energy required to split an atom into its component particles, it is called the binding energy of the particular atom. The relationship between binding energy and mass defect is given by Einstein's equation, $E = mc^2$.

The binding energy varies from atom to atom and is usually expressed in terms of the binding energy per nucleon in a particular atom. The chart above shows a plot of the binding energy per nucleon against the mass number (A) for the most stable atoms in nature.

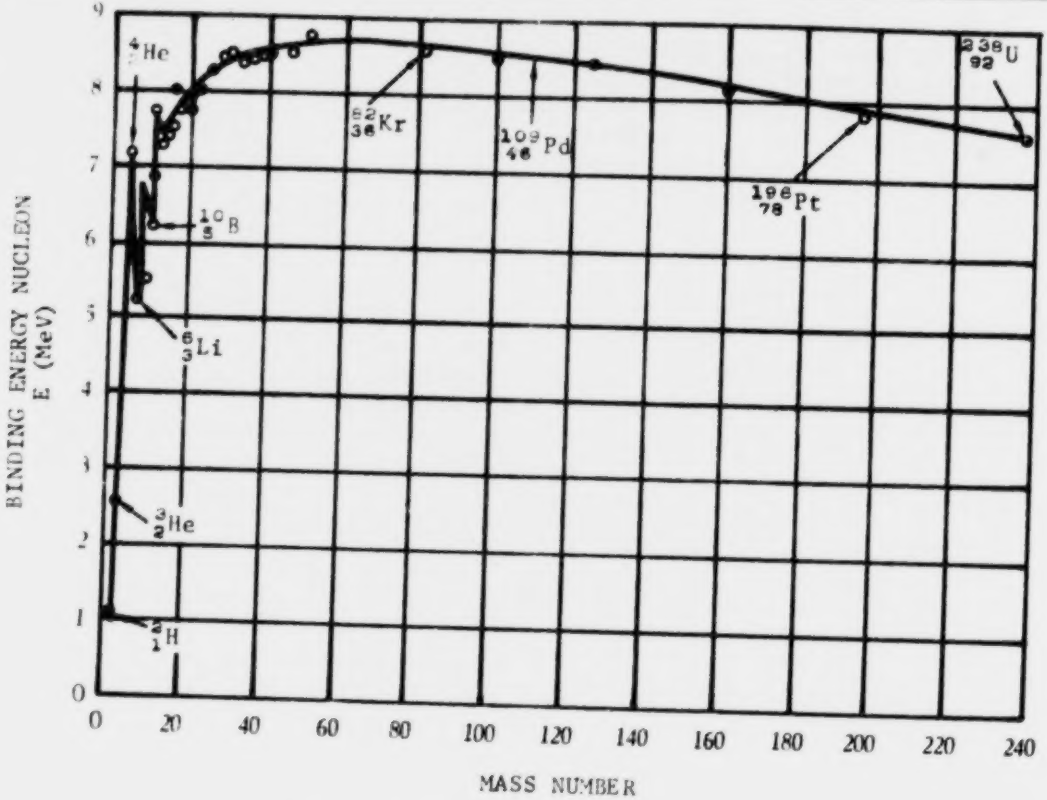

Figure 4. Binding Energy vs. Mass Number

Important features of Figure 4 are (1) a central region for which the average binding energy is nearly constant (≅8.5 MeV/nucleons); (2) regions at either end of the curve featuring a smaller binding energy per particle (which means a less stable nucleus for both small and large mass numbers); and (3) an unusually large binding energy for small mass numbers having nucleons in multiples of four.

Consider the formation of deuterium as a simple example of the meaning of binding energy. Deuterium is made up of one proton and one neutron in the nucleus, and one electron outside. The atomic mass of deuterium is 2.01410 amu. The sum of the masses of its component particles, however, is:

mass of proton =	1.00728
mass of neutron =	1.00866
mass of electron =	.00055
sum	2.01649

Thus, the mass of the deuterium atom is less than the mass of its component particles by an amount of .00239 amu.

Since 1 amu is equivalent to 931 MeV, the binding energy of the deuterium atom may now be written as:

$$E = .00239 \times 931 \text{ MeV} = 2.23 \text{ MeV}$$

The binding energy of the electron is approximately 14 eV and is, therefore, negligible in comparison with 2.23 MeV. Thus, the entire binding energy is essentially that of the two nucleons in the nucleus of the deuterium atom or about 1.11 MeV each.

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Radioactivity

I. INTRODUCTION

The atom is comprised of a nucleus, consisting of neutrons and protons, and orbital electrons. The atomic number (Z) and the mass number (A) describe the quantity of these atomic components. There are 92 naturally occurring elements with atomic numbers ranging from 1 to 92. The atomic masses are approximately integral numbers that increase roughly in proportion to the atomic numbers. All elements have several isotopes, differing in mass by one or more mass units from each other. The integral nature of the increments, both in atomic number and mass number, suggests that the nucleus itself is a structure of particles (protons and neutrons) with about one atomic mass unit each. The protons carry one unit positive charge; the neutrons are neutral.

Certain structural criteria must be satisfied for a nucleus to exist in a stable state. When these criteria are not met, the nucleus will be unstable and manifest the phenomenon of radioactivity. Radioactivity refers to the processes by which a nucleus decays--or disintegrates--by one or more discrete energy steps or transitions, until a stable state is reached. The criteria which determine a nucleus's stability or instability are briefly discussed in the following section.

II. NUCLEAR STABILITY

There are two sets of forces acting on the particles within the nucleus: the relatively long-range coulomb forces of repulsion between the positive charges on the protons, and the very short-range forces between both types of particles. These latter forces are only partly understood; no attempt will be made here to discuss them. Suffice to say the combined effects of these attractive and repulsive forces are such that only certain combinations or ratios of neutrons and protons are stable.

Theoretically, if the coulomb forces were discounted, optimum stability would be achieved within a nucleus when the numbers of protons and neutrons are approximately equal. Such is the situation for the lighter elements in which the coulomb forces, relative to the short-range forces, are insignificant. These coulomb forces become increasingly significant as the atomic number increases above 20. With large increases of these repulsive forces within nuclei, intranuclear stability conditions become altered. Consequently, with increasing atomic number a neutron excess is required for a stable nuclide.

Thus, for elements in the Periodic Table with low atomic numbers, greater nuclear stability is found when the number of neutrons is about equal to the number of protons; stated in another manner, when the ratio of neutrons to protons (n:p ratio) is approximately equal to unity. As elements increase in Z number above 20, the n:p ratio required for stability gradually increases until Z = 83 (bismuth), where the n:p ratio required for stability exceeds 1.5:1. Finally, at the high atomic number part of the Periodic Table, above $^{209}_{83}$Bi, there are no completely stable nuclei. The stability conditions based on n:p ratios are not very critical, and "stability ranges" of n:p ratios exist for any given atomic number or mass number. For a given mass number, there may be several stable arrangements of protons and neutrons resulting in several stable isobars. For a given atomic number, conditions may vary still more widely, so numerous stable isotopes can occur for a particular element--as many as 10 stable isotopes for some elements.

The n:p ratio is an oversimplified explanation of the intranuclear criteria leading to nuclear stability. Additional factors involving quantum-mechanical principles and binding energy considerations are needed to fully explain nuclear stability. These are beyond the scope of this presentation.

In summary, nuclear stability is governed by the particular combination and arrangement of neutrons and protons in a given nucleus. If the combination and arrangement of neutrons and protons does not fall within a "stable range," the nucleus is unstable. This is tantamount to saying the nucleus is radioactive. An unstable nucleus attempts to achieve stability by changing its configuration or ratio of neutrons and protons by means of spontaneous disintegration, or, radioactive decay.

III. RADIOACTIVITY

For the purpose of this discussion, a distinction will be made between natural radioactivity and artificial radioactivity.

A. Natural and Artificial

Certain nuclides are unstable as they occur in nature and are therefore called "natural" radionuclides.

Others occur as a result of various man-made reactions. The vast majority of radionuclides are produced in this manner. These are called "artificial" radionuclides. The processes involved in the production of artificial radioactivity will be discussed in detail later.

B. History

Evidence of natural radioactivity was first reported by Henry Becquerel in 1896. Becquerel demonstrated that uranium ore would fog or darken a photographic plate shielded with opaque paper in much the same manner as x rays. He postulated that the uranium emitted very penetrating rays, similar to x rays. The phenomenon ultimately was called "radioactivity." In time, it was determined that there were many elements beyond the atomic number of lead (Z = 82) which showed similar radiating characteristics.

After a long and complicated chain of investigations, to which many outstanding physicists contributed, a better understanding of natural radioactivity was available. The understanding culminated with the experiments of Rutherford. In 1903, he clearly showed there were three kinds of radioactive emissions, namely: alpha, beta, and gamma.

C. Types of Radioactive Emissions

Alpha particles are the least penetrating of the three types of radiation. They can be absorbed or stopped by a few centimeters of air or a thin piece of paper. They have an electric charge opposite to and exactly twice that of the electron, and a mass number of 4. Thus, an alpha particle and a helium nucleus are identical in structure. With a few exceptions, only relatively heavy radioactive nuclides decay by alpha emission.

Beta particles are negatively charged particles. They have the same mass and charge as an electron and can be considered high-speed electrons. They originate in the nucleus, in contrast with ordinary electrons, which exist in the orbits around the nucleus. They travel several hundred times the distance of alpha particles in air and require a few millimeters of aluminum to stop them. Beta activity can be expected where the n:p ratio is high. The intranuclear effect of a beta emission would be changing a neutron into a proton, decreasing the n:p ratio.

Gamma rays are a type of electromagnetic radiation and move with the speed of light. The basic difference between gamma rays and x rays is their source; gamma rays originate in the nucleus, x rays in the orbital electron structure. The basic difference between gamma rays and visible light is their frequency. The Greek symbol γ is used to designate gamma radiation.

In early investigations of radioactivity, all three of the previously mentioned radiations were commonly referred to as "rays," establishing a terminology still in use despite the fact that the term "ray" is a misnomer for particulate emissions.

Other types of emissions. There are several other types of radioactive emissions usually associated with artificial radiactivity. Because of characteristics in common with alpha, beta, and gamma rays, they are described in this section.

1. Positron particles are similar to beta particles except that they are positively charged. Thus, the positron can be considered a positively charged, high-speed electron which originates in the nucleus.

2. Orbital electron capture (K-capture) refers to the radioactive decay process whereby the nucleus captures an electron from an orbital shell of the atom. An electron from a higher energy level immediately moves in to fill the vacancy, and the excess energy is emitted as a characteristic x ray photon. The nucleus might conceivably capture an L-shell electron, but K-electron capture is much more probable. This mode of decay is frequently referred to as "K-capture." Orbital electron capture is generally abbreviated as "E.C."

Both positron emission and K-capture can be expected in nuclides with a low n:p ratio. The intranuclear effect of either mode of decay would be to change a proton into a neutron, thus increasing the n:p ratio.

3. Internal conversion occurs when a gamma photon does not escape the electron cloud surrounding the nucleus, but transfers to one of the orbital electrons enough energy to eject it from the atom. This process usually takes place in the K-shell. In principle, it is similar to the photoelectric effect.

4. Isomeric transition is a type of radioactive decay associated with certain pairs of nuclides that have the same mass number and the same atomic number, but different energy contents. Generally, one of the isomers of the pair is metastable and achieves ground state or a lower energy state by emitting delayed (usually greater than 10^{-9} seconds) gamma radiation. The metastable or excited state, therefore, represents one isomeric form of a particular nuclear species while the ground state represents the other. Generally, further radioactive decay by beta emission occurs from the ground state of the parent nuclide.

5. Neutrinos are neutral particles with negligible rest mass. They account for the energy distribution among beta particles in the beta-decay process from a given radionuclide.

D. Transmutation

When a radioactive nuclide decays by **α**, **β^-**, **β^+** or K-capture, a transmutation occurs. The decay product--or "daughter"--has become an atom of a new element with chemical properties entirely unlike the original "parent" atom.

A nucleus emitting an alpha particle disintegrates to a daughter element, reduced in atomic number by 2 and reduced in atomic mass by 4; e.g., radium decays by alpha emission to produce radon as follows:

$${}^{226}_{88}\mathrm{Ra} \longrightarrow {}^{222}_{86}\mathrm{Rn} + {}^{4}_{2}\mathrm{He}$$

In beta emitters, the nucleus of the parent gives off a negatively charged particle, resulting in a daughter more positive by one unit of charge. The atomic number increases by one, but the mass number is unchanged, e.g., lead decays by beta emission to produce bismuth as follows:

$${}^{214}_{82}\mathrm{Pb} \longrightarrow {}^{214}_{83}\mathrm{Bi} + {}^{0}_{-1}\mathrm{e}$$

With positron emitters, the parent nucleus gives off a positive charge. This results in a daughter less positive by one unit of charge. The atomic number decreases by one and the mass number remains unchanged, as in the following reaction:

$${}^{57}_{28}\mathrm{Ni} \longrightarrow {}^{57}_{27}\mathrm{Co} + {}^{0}_{1}\mathrm{e}$$

If K-capture is the mode of decay, the transmutation resembles that of positron emission, as follows:

$${}^{57}_{28}\mathrm{Ni} + {}^{0}_{1}\mathrm{e} \longrightarrow {}^{57}_{27}\mathrm{Co}$$

Note that ${}^{57}\mathrm{Ni}$ has two modes of decay. This is an example of the "branching" which is explained in the next section, Decay Phenomena.

Nuclear decay reactions resulting in a transmutation generally leave the resultant nucleus in an excited state. Nuclei, thus excited, may reach an unexcited or ground state by instantaneous emission of a gamma photon. All of the transmutation examples given would be accompanied by gamma emission. Although most nuclear decay reactions do have gamma emissions associated with them, there are numerous radionuclide species which decay by particulate emission with no gamma emission.

E. Decay Phenomena

Each radioactive nuclide, artificial and natural, has its own unique characteristic pattern of decay. There are two aspects associated with this "pattern"; one is the particular types and energies of the emissions involved, and the other is the rate of decay.

All nuclei of a given nuclide seeking stability by radioactive decay do so in a specific manner. As indicated, ^{226}Ra decays by an alpha emission which is accompanied by a gamma photon. This represents the only mode of decay open to ^{226}Ra. There are some radioactive nuclides which may decay with branching, whereby a choice of decay modes exists. In such case, a definite branching ratio exists. A case in point is the decay of ^{57}Ni. This isotope of nickel decays 50% by K-capture and 50% by β^+ emission. The branching ratio would be

$$\frac{\beta^+}{\mathrm{EC}} = 1$$

Not only do various radionuclides disintegrate in a constant manner insofar as the types of emissions are concerned, but the emissions from each nuclide exhibit a distinct energy picture. The energies associated with radiations are given in terms of "million electron volts" (MeV). Beta emissions may occur with energies to about 5 MeV; alphas to about 10 MeV; and gamma photons to about 3 MeV. The energy of the particulate radiations is manifested as kinetic energy--the higher the energy the greater the velocity of the particle. However, the velocity of photons is constant (c = speed of light) and energy differences are manifested by varying wavelengths and frequencies.

The other characteristic aspect associated with decay patterns is the rate of decay. The disintegrations of radionuclides occur with a regularity characteristic for each particular species. Such disintegrations are spontaneous and random. A single radium nucleus, for instance, may disintegrate at once or wait thousands of years before emitting an alpha particle. All that can be predicted with any certainty is that half of all the ^{226}Ra nuclei present will disintegrate in 1622 years. This period is called the half-life of ^{226}Ra.

The wide range in half-life values for naturally occurring radionuclides is typified by ^{212}Po, with a half-life of 3×10^{-7} seconds, and ^{209}Bi, with a half-life of over 2×10^{18} years.

F. Radioactive Families

The transmutations associated with naturally occurring radionuclides frequently yield a daughter which also is radioactive. To date, about 70 different naturally occurring radionuclides have been identified, each with its own characteristic pattern of radioactivity. Most of these yield radioactive daughters and are now known to be intimately interrelated in radioactive series or families. It has been established that most isolated radioactive species belong to one of three independent groups or families. Each family starts with a parent radionuclide, decaying or transmuting into a radioactive daughter nuclide, which would again transmute into a daughter nuclide, also radioactive, and so on until stability is attained. One family starts with uranium-238 and is called the uranium series. Another starts with thorium-232 and is called the thorium series. A third starts with uranium-235 and is called the actinium series. Summaries of these series are found in Tables 1-3. In each series, there is a "seesawing" in the transmutation chain between decreasing the atomic number by two with α emission and increasing it by one with β emission. Each series ends in a different stable isotope of lead. There is also a fourth series, the neptunium series, named after its longest-lived member. (See Table 4.) Actually, the neptunium series has been artificially produced and does not occur in nature, but it is assumed that it did occur in nature at one time and has become extinct because of the relatively short half-lives involved. The longest-lived radionuclide in the series is ${}^{237}_{93}\text{Np}$ with a half-life of 2.2×10^6 years. Assuming the age of the earth is 2.2×10^9 years, this would indicate that, from the time of creation, ${}^{237}\text{Np}$ has undergone 1000 half-lives decay.

The fraction of a radionuclide remaining after 1000 half-lives would be fantastically small--in the order of 10^{-300}. It is obvious, therefore, why it would be difficult to find traces of neptunium and its descendants in nature.

G. Singly Occurring Radionuclides

Careful measurements show that almost all materials contain traces of radioactivity. One might suspect that these traces might be due to contamination with some of the heavy radionuclides belonging to one of the radioactive series described. However, some of the lighter elements are themselves weakly radioactive. Table 5 lists five examples of naturally occurring radionuclides and their radiations.

Currently, there is no evidence that any series relationship exists among these nuclides. Therefore, they are regarded as individual, naturally occurring radionuclides with no familial relationships.

IV. ARTIFICIAL RADIOACTIVITY

As implied in the nomenclature, natural and artificial radioactivity differ in origin. There are other distinctions between the two types which will be discussed. Nevertheless, the nuclei of artificial radionuclides are unstable in much the same manner as their natural counterparts. The intranuclear factors governing decay are also similar for both groups.

A brief account of the discovery of artificial radioactivity will be given before further discussing its similarities and dissimilarities to natural radioactivity.

A. Induced Transmutations

In 1919, Lord Rutherford demonstrated that it was possible to produce artificially a transmutation of elements. How naturally occurring unstable or radioactive atoms are transmuted by ejecting ionizing particles and rays has been discussed. Lord Rutherford set up and observed a nuclear reaction in reverse, one might say, whereby high-speed charged particles (projectiles) bombarded stable atomic nuclei (target), resulting in a reaction at the nuclear level and inducing a transmutation. The first observed nuclear reaction used alpha particles from ${}^{214}\text{Bi}$ (RaC) as the charged particles. These were made to impinge upon nitrogen nuclei, acting as the target. The reaction is written as follows:

$${}^{14}_{7}\text{N} + {}^{4}_{2}\text{He} \longrightarrow {}^{17}_{8}\text{O} + {}^{1}_{1}\text{H}$$

Since the discovery of the transmutation of nitrogen, many hundreds of artificial or induced transmutations have been found. Until 1932, most induced transmutations were performed utilizing naturally occurring alpha emitters as sources of incident particles. With the development of particle accelerators, other bombarding particles have been successfully used. No attempt will be made here to catalog the many kinds of possible transmutations, nor will any attempt be made to discuss the theory and quantitative data regarding nuclear reactions.

B. Induced Radioactivity

During the first 15 years of experimental work with nuclear reactions, the transmutation products (insofar as could be observed)

TABLE 1.--The Uranium Series

Radioelement	Corresponding Element	Symbol	Radiation	Half Life
Uranium I	Uranium	^{238}U	α	4.51 x 10^9 y
Uranium X_1	Thorium	^{234}Th	β	24.1 d
Uranium X_2*	Protactinium	^{234}Pa	β	1.18 m
Uranium II	Uranium	^{234}U	α	2.48 x 10^5 y
Ionium	Thorium	^{230}Th	α	8.0 x 10^4 y
Radium	Radium	^{226}Ra	α	1.62 x 10^3 y
Ra Emanation	Radon	^{222}Rn	α	3.82 d
Radium A 99.98% / 0.02%	Polonium	^{218}Po	α and β	3.05 m
Radium B	Lead	^{214}Pb	β	26.8 m
Astatine-218	Astatine	^{218}At	α	2 s
Radium C 99.96% / 0.04%	Bismuth	^{214}Bi	β and α	19.7 m
Radium C'	Polonium	^{214}Po	α	1.6 x 10^{-4} s
Radium C''	Thallium	^{210}Tl	β	1.32 m
Radium D	Lead	^{210}Pb	β	19.4 y
Radium E 100% / 2 x 10^{-4}%	Bismuth	^{210}Bi	β and α	5.0 d
Radium F	Polonium	^{210}Po	α	138.4 d
Thallium-206	Thallium	^{206}Tl	β	4.20 m
Radium G (End Product)	Lead	^{206}Pb	Stable Stable	--

*Uranium X_2 undergoes isomeric transition to form uranium Z (^{234}Pa); the latter has a half-life of 6.7 h, emitting β radiation and forming uranium II (^{234}U).

TABLE 3.--The Actinium Series

Radioelement	Corresponding Element	Symbol	Radiation	Half Life
Actinouranium	Uranium	^{235}U	α	7.1 x 10^8 y
Uranium Y	Thorium	^{231}Th	β	25.5 h
Protactinium	Protactinium	^{231}Pa	α	3.43 x 10^4 y
Actinium 98.8% / 1.2%	Actinium	^{227}Ac	β and α	22 y
Radioactinium	Thorium	^{227}Th	α	18.6 d
Actinium K	Francium	^{223}Fr	β	22 m
Actinium X	Radium	^{223}Ra	α	11.4 d
Ac Emanation	Radon	^{219}Rn	α	3.92 s
Actinium A ~100% / 5 x 10^{-4}%	Polonium	^{215}Po	α and β	1.83 x 10^{-3} s
Actinium B	Lead	^{211}Pb	β	36.1 m
Astatine-215	Astatine	^{215}At	α	~10^{-4} s
Actinium C 99.68% / 0.32%	Bismuth	^{211}Bi	α and β	2.16 m
Actinium C'	Polonium	^{211}Po	α	0.52 s
Actinium C''	Thallium	^{207}Tl	β	4.79 m
Actinium D (End Product)	Lead	^{207}Pb	Stable	--

TABLE 2.--The Thorium Series

Radioelement	Corresponding Element	Symbol	Radiation	Half Life
Thorium	Thorium	^{232}Th	α	1.39 x 10^{10} y
Mesothorium I	Radium	^{228}Ra	β	6.7 y
Mesothorium II	Actinium	^{228}Ac	β	6.13 h
Radiothorium	Thorium	^{228}Th	α	1.91 y
Thorium X	Radium	^{224}Ra	α	3.64 d
Th Emanation	Radon	^{220}Rn	α	52 s
Thorium A	Polonium	^{216}Po	α	0.16 s
Thorium B	Lead	^{212}Pb	β	10.6 h
Thorium C 66.3% / 33.7%	Bismuth	^{212}Bi	β and α	60.5 m
Thorium C'	Polonium	^{212}Po	α	3 x 10^{-7} s
Thorium C''	Thallium	^{208}Tl	β	3.1 m
Thorium D (End Product)	Lead	^{208}Pb	Stable	—

TABLE 4.--The Neptunium Series

Element	Symbol	Radiation	Half Life
Plutonium	^{241}Pu	β	13.2 y
Americium	^{241}Am	α	462 y
Neptunium	^{237}Np	α	2.20 x 10^6 y
Protactinium	^{233}Pa	β	27.4 d
Uranium	^{233}U	α	1.62 x 10^5 y
Thorium	^{229}Th	α	7.34 x 10^3 y
Radium	^{225}Ra	β	14.8 d
Actinium	^{225}Ac	α	10.0 d
Francium	^{221}Fr	α	4.8 m
Astatine	^{217}At	α	1.8 x 10^{-2} s
Bismuth 98% / 2%	^{213}Bi	β and α	47 m
Polonium	^{213}Po	α	4.2 x 10^{-6} s
Thallium	^{209}Tl	β	2.2 m
Lead	^{209}Pb	β	3.32 h
Bismuth (End Product)	^{209}Bi	Stable	—

TABLE 5.--Naturally Occurring Radionuclides

Nuclide	Half-Life	Emission Energies in MeV		
		Alpha	Beta	Gamma
$^{40}_{19}K$	1.26×10^{9} y	---	1.314	1.460
$^{87}_{37}Rb$	5.0×10^{10} y	---	0.274	---
$^{147}_{62}Sm$	1.05×10^{11} y	2.23	---	---
$^{176}_{71}Lu$	3.0×10^{10} y	---	0.43	.088 .202 .306
$^{187}_{75}Re$	4.0×10^{10} y	---	.003	.134

were not radioactive. However, the reactions generally were accompanied by the emission of a charged particle and a gamma ray. These emissions are not construed as imparting the property of radioactivity to the target element, since they occur practically instantaneously.

It was determined in 1934 that induced transmutations could produce nuclei which were residually unstable in somewhat the same manner as naturally occurring radionuclides. Madame Curie and M. Joliot reported that certain light elements (boron, magnesium, aluminum), when bombarded with alpha particles continued to emit radiation for a finite time after bombardment had stopped. The following reaction, involving aluminum bombarded with alpha particles, was the first reported instance of induced or artificial radioactivity:

$$^{27}_{13}Al + ^{4}_{2}He \longrightarrow ^{30}_{15}P + ^{1}_{0}n$$

The resultant nucleus $^{30}_{15}P$ was observed to be radioactive, decaying by β^{+} emission with a half-life of 2.6 minutes.

The work of Curie and Joliot stimulated similar experiments throughout the world. As a result, radioactive isotopes of nearly every element in the Periodic Table were produced by bombarding a stable isotope with charged particles, neutrons, or, in certain instances, photons. Over 1,000 unstable nuclear species are listed in the Table of Isotopes(1).

It may have been noted that carbon-14 was not included as a natural radionuclide even though it has received considerable popular attention in recent years as "naturally" occurring elemental carbon in definite, though small, proportions. The ^{14}C existing in the atmosphere is being formed continually as a result of nuclear reactions between atmospheric nitrogen and neutrons from cosmic rays. This is shown in the following reaction:

$$^{14}_{7}N + ^{1}_{0}n \longrightarrow ^{14}_{6}C + ^{1}_{1}H$$

Naturally occurring radionuclides, in becoming stable, generally emit beta or alpha particles by a series of transmutations. Artificial radionuclides rarely decay by alpha emission; they usually decay by β^{-}, β^{+} emission or orbital electron capture. In contrast to natural radioactivity, induced radionuclides generally revert to stability in a single decay step.

C. Fission Products

Another source of radionuclides is nuclear fission. The nuclear fragments directly resulting from fission invariably have too large a proportion of neutrons to protons for stability, and consequently tend to achieve stability by negative beta emission. Each nuclear fragment initiates a radioactive series (fission decay chain) involving the successive emission of beta particles. Fission product beta emission, as with other beta emitters, generally is accompanied by gamma emission.

REFERENCE

(1) Lederer, C. M., Hollander, J. M., and Perlman, I., Table of Isotopes (6th ed.; New York: John Wiley & Sons, Inc., 1967).

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Chart of the Nuclides

I. INTRODUCTION

The nuclei of atoms are composed of protons and neutrons. These basic atomic building blocks are collectively referred to as nucleons.

A. Element--the number of protons in the nucleus determines the element. For example, all atoms whose nuclei contain 6 protons are the element carbon, those containing 38 protons are strontium, and those containing 92 protons are uranium.

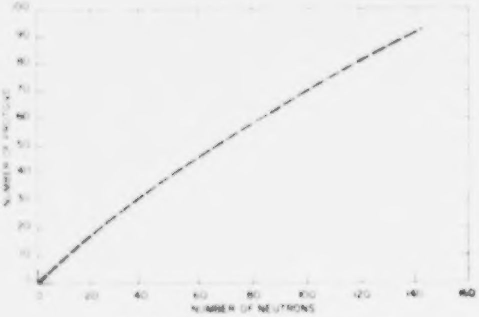
B. Isotope--all atoms of one element do not necessarily have the same number of neutrons. Atoms of the same element which differ in the number of neutrons in their nuclei are called isotopes.

C. Nuclei--a given element may have one or more isotopes. The isotopes of all the elements are known as nuclides and a chart listing all the isotopes of all the elements is called a chart of the nuclides.

II. GENERAL ARRANGEMENT

A. In arranging the nuclides in chart form, usually the number of protons (or atomic number Z) is plotted vertically and the number of neutrons (N) horizontally. (See Figure 1).

B. Such a plot at once reveals the continuity in composition in progressing from the lighter to the heavier elements.



THE ELEMENTS AND THEIR ISOTOPES

Figure 1

III. SPECIFIC NUCLIDE REPRESENTATION

A. Each specific nuclide is represented on the Chart of the Nuclides by a block. A typical section of the Chart is shown in Figure 2.

B. The coloring and labeling of each block specifies certain information concerning the properties of the nuclide.

1. A grey block denotes a stable nuclide. A typical example, stable sodium, with a key to the listed data within the block is shown in Figure 3.

2. Unlike sodium, most elements have more than one stable isotope. For example, manganesium has three stable isotopes as shown in Figure 4.

3. A white block denotes an artificially produced radioactive nuclide. A typical

example with a key to data listed within the block is shown in Figure 7.

4. A white block with black corner denotes an artificially produced radionuclide resulting from fission. A typical example of this type nuclide is shown in Figure 6.

5. Certain nuclides, even though naturally occurring, are radioactive. These are denoted by a grey block with a black border at the top. (See Figure 5).

IV. DEPICTING NUCLEAR PROCESSES

A. As a result of decay, radionuclides shift from block to block within the Chart of the Nuclides. The relative locations of the products of various nuclear processes are shown in the left hand portion of Figure 8.

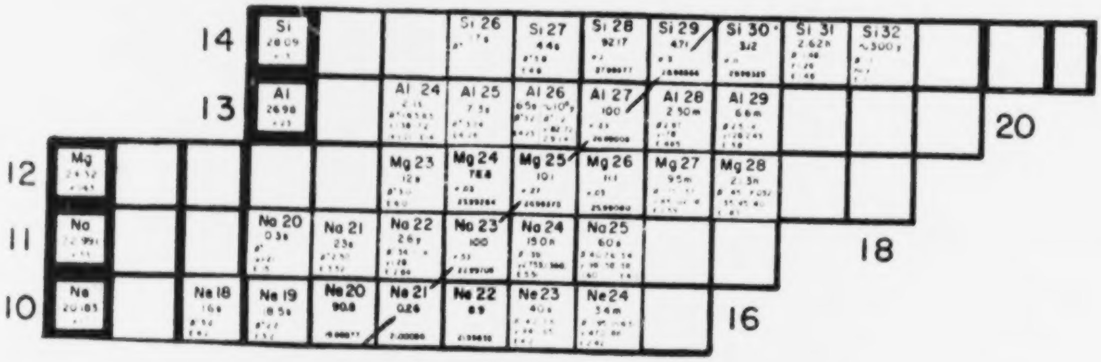
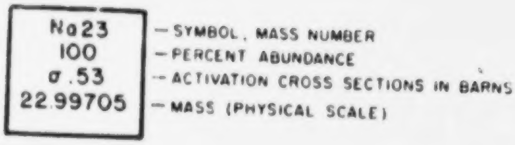
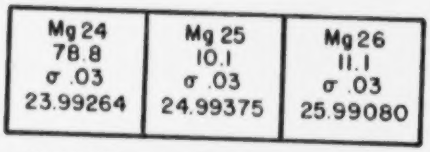

14
13
12
11
10
20
18
16
Si
Si 26
Si 27
Si 28
Si 29
Si 30
Si 31
Si 32
Al
Al 24
Al 25
Al 26
Al 27
Al 28
Al 29
Mg
Mg 23
Mg 24
Mg 25
Mg 26
Mg 27
Mg 28
Na
Na 20
Na 21
Na 22
Na 23
Na 24
Na 25
Ne
Ne 18
Ne 19
Ne 20
Ne 21
Ne 22
Ne 23
Ne 24

Figure 2


Na 23
100
σ .53
22.99705
— SYMBOL, MASS NUMBER
— PERCENT ABUNDANCE
— ACTIVATION CROSS SECTIONS IN BARNS
— MASS (PHYSICAL SCALE)

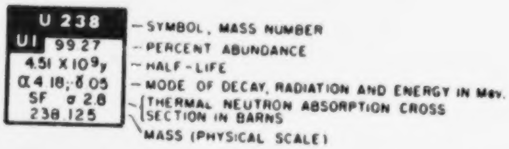
A GREY BLOCK DENOTES
A STABLE NUCLIDE

Figure 3


Mg 24
78.8
σ .03
23.99264
Mg 25
10.1
σ .03
24.99375
Mg 26
11.1
σ .03
25.99080

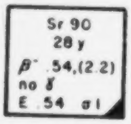
MANY ELEMENTS HAVE MORE
THAN ONE STABLE ISOTOPE

Figure 4


U 238
UI 99.27
4.51 X 10⁹y
α 4.18; γ .05
SF σ 2.8
238.125
— SYMBOL, MASS NUMBER
— PERCENT ABUNDANCE
— HALF-LIFE
— MODE OF DECAY, RADIATION AND ENERGY IN Mev.
THERMAL NEUTRON ABSORPTION CROSS SECTION IN BARNS
MASS (PHYSICAL SCALE)

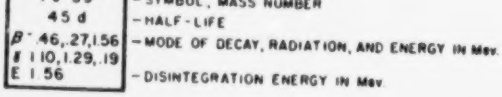
A GREY BLOCK WITH BLACK BAND DENOTES
A NATURALLY OCCURRING RADIONUCLIDE

Figure 5


Sr 90
28 y
β⁻ .54,(2.2)
no γ
E .54 σ 1

A BLOCK WITH BLACK CORNER
DENOTES A FISSION PRODUCT

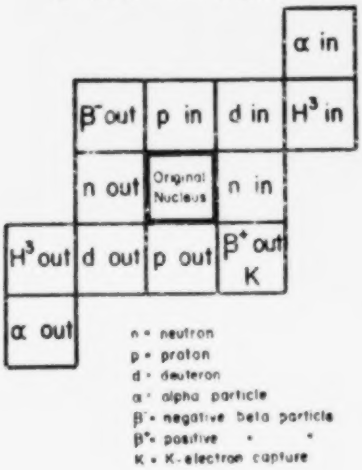
Figure 6


Fe 59
45 d
β⁻ .46,.27,1.56
γ 1.10,1.29,.19
E 1.56
— SYMBOL, MASS NUMBER
— HALF-LIFE
— MODE OF DECAY, RADIATION, AND ENERGY IN Mev.
— DISINTEGRATION ENERGY IN Mev.

A WHITE BLOCK DENOTES AN ARTIFICIALLY
PRODUCED RADIONUCLIDE

Figure 7

Relative Locations of the Products of Various Nuclear Processes


α in
β⁻ out
p in
d in
H³ in
n out
Original Nucleus
n in
H³ out
d out
p out
β⁺ out K
α out
n = neutron
p = proton
d = deuteron
α = alpha particle
β⁻ = negative beta particle
β⁺ = positive " "
K = K-electron capture

Displacements caused by Nuclear Bombardment Reactions

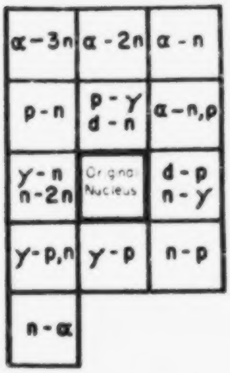

α-3n
α-2n
α-n
p-n
p-γ d-n
α-n,p
γ-n n-2n
Original Nucleus
d-p n-γ
γ-p,n
γ-p
n-p
n-α

Figure 8

B. Displacements also occur as a result of nuclear reactions brought about through bombarding given nuclides with various nuclear particles or gamma photons. These changes are depicted in the lower portion of Figure 8.

V. COMMENTARY

The Chart of the Nuclides provides considerable information about the behavior of nuclides.

A. As mentioned previously, there is continuity in composition of the nuclides. For example, as shown in Figure 1, a line drawn through the stable nuclides forms a rather smooth curve extending from the lower left to the upper right corner of the Chart of the Nuclides.

1. Nuclides below this line are characterized by having an excess of neutrons and will in general be beta particle emitters.

2. Nuclides lying above the line are characterized by having an excess of protons and will in general decay by positron emission or K-capture.

B. The Chart presents in compact style much valuable information concerning the properties of the nuclides. These data include for:

1. STABLE NUCLIDES

- a. Relative Abundance
- b. Cross Section for Activation

2. RADIOACTIVE NUCLIDES

- a. Type of Emissions
- b. Energy of Emissions
- c. Half Life

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2. "A Supplement to the Charts on Nuclear Physics", Prepared by Westinghouse School Service in cooperation with scientists of the Westinghouse Research Laboratories, Westinghouse Electric Corp., 306 Fourth Ave., P. O. Box 1017, Pittsburgh 30, Pennsylvania - 1948, Price 10¢.

Quantities and Units of Radiation

I. INTRODUCTION

The structure and meaning of currently accepted definitions used in quantitative determination of radiation fields must be understood. Without a physically consistent, realistic, and measurable description of the radiation field, the clinician, radiologist, or technologist could not utilize radiation with reproducibility or safety, nor could the effects of radiation on living organisms be ascertained. The concepts discussed in this outline represent only part of a set of radiation quantities and units established by the International Commission on Radiation Units and Measurements (ICRU) in its Report Number 11, Radiation Quantities and Units, published in 1968.

II. QUANTITIES AND UNITS

The ICRU has standards which clearly delineate radiation quantities and radiation units.

A "quantity" may be thought of as a description of a physical concept or principle. The magnitude or measure of a quantity is a "unit." Fundamentally, therefore, the quantity is the more important. The unit places limits upon, but will not serve to define, the quantity. For example, the concept of length is a quantity; the meter is a unit of length.

An internationally accepted system of writing and abbreviating units has been established. To avoid confusion with the name of the man for whom a unit may be named, the name is capitalized when referring to the man (Roentgen) but when referring to the unit, the name will be written in lower case (roentgen).

The abbreviation for the unit is, however, capitalized (R). In cases where the abbreviation conflicts with another accepted symbol, a combination of letters, usually the first and last, is used. An illustration is seen in the case of the unit curie. The abbreviation used is Ci, as other possible symbols conflict with the symbol for coulomb and with common chemical symbols.

A. Radiation Quantities

Radiation may be divided into two classes: ionizing and non-ionizing. X rays, for example, are classified as ionizing radiations by virtue of their energy. Ionizing radiations, however, include many types of radiations in addition to x rays. The ICRU has considered ionizing radiations in two categories:

1. Directly ionizing particles or charged particles (electrons, protons, alpha particles, etc.), which have sufficient kinetic energy to cause ionization by collision; and
2. Indirectly ionizing particles or uncharged particles (neutrons, gamma rays, x rays, etc.), which can interact and liberate directly ionizing particles.

Quantities describing the effects of the interaction of directly and indirectly ionizing radiations with matter have been developed to cover many aspects of the interaction processes. For practical radiation protection measurements and control, only four quantities are generally utilized. These will be discussed in greater detail in the following material.

B. Exposure

Historically, the most important term associated with radiation exposure has been the roentgen. So, after a series of redefinitions, the roentgen has been chosen as the unit for the quantity exposure. The current definition of exposure is:

> The exposure (X) is the quotient of ΔQ by Δm, where ΔQ is the sum of the electrical charges on all the ions of one sign produced in air when all the electrons (negatrons and positrons), liberated by photons in a volume element of air whose mass is Δm, are completely stopped in air.

$$X = \frac{\Delta Q}{\Delta m}.$$

Several important restrictions applying to the quantity are immediately apparent from the wording of the definition:

1. The medium in which exposure is defined is air; no other medium is acceptable.
2. The radiations for which exposure is defined are photons (x ray and gamma).
3. The effect being measured is ionization of the air.
4. All electrons liberated in the ionizing processes must be stopped in air.

The last restriction serves to impose upper limits on the maximum quantum energy which may be directly measured in terms of exposure, because the range of the electrons liberated increases as a function of the energy. A chamber (see a discussion of the free air chamber) which measures exposure directly must have dimensions such that the distances from the electrodes to the sites of interaction are greater than the maximum range of the secondary electrons. For example, a chamber intended to measure exposure resulting from a beam of 2 MeV photons would require overall dimensions greater than 15 meters, because the maximum range in air of 2 MeV electrons which may be liberated is on the order of 7.5 meters.

The meaning of the symbol Δ (read delta) is best illustrated by considering the fact that the interaction of radiation is not continuous. Interaction locations are discrete sites which can only be described statistically. If the measuring volume is too small, a single measurement of the effect may include no interaction or, through chance, several interactions may have occurred within the volume. Therefore, an average value must be determined. This effect of random variations obtained by a series of measurements is shown in the left region of the curve in Figure 1. As the volume is increased, the effect of the larger volume is to average the effects measured in a series of smaller volumes. The result approaches closely the true average, as indicated in the center portion of the curve. A continued increase in the volume would now include regions outside the radiation field, and the measured values would decrease, as shown on the right. In short, the measurement of radiation fields involves statistical averaging procedures; the symbol Δ will precede the symbols for quantities that may be involved in such averaging procedures.

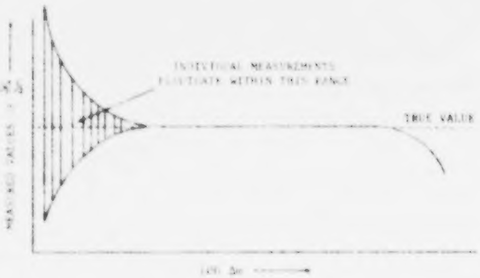

Figure 1.--Detector Response vs. Volume

C. Absorbed Dose

Exposure, as noted earlier, is limited to measurement of the ionization in air produced by x or γ radiations. This quantity is, of course, extremely limited: Many types of radiation exist, and the material of interest in which interaction occurs in most cases is not air. Additionally, ionization produced by the radiation may not be the effect which is most usefully evaluated. "Absorbed dose" is a quantity much more general in its structuring. It is defined as follows:

> The absorbed dose (D) is the quotient of ΔE_D by Δm, where ΔE_D is the energy imparted

by ionizing radiation to the matter in a volume element, and Δm is the mass of the matter in that volume element.

$$D = \frac{\Delta E_D}{\Delta m}.$$

The following points are to be seen in the definition:

1. Any type of ionizing radiation (directly or indirectly ionizing) is included.
2. Any material may be used in making the measurement.
3. The effect observed is the energy imparted by the radiation to the matter in the volume element.

Most often a direct measurement of absorbed dose will be carried out by calorimetry, since temperature changes--which directly reflect the energy input to the mass (if one excludes possible changes from associated chemical energies)--are relatively easy to measure. Again, it must be pointed out that selection of the size of the mass is important as statistical averaging procedures apply.

The energy imparted to the matter in the volume element is not necessarily equal to the amount of energy removed from the beam. Secondary electrons, for example, may carry some energy to matter in adjacent volumes. The energy lost is related to another quantity, "kerma" (kinetic energy released in material).

Because absorbed dose is a general quantity applying to any radiation in any material, it could certainly be applied to x radiation in air. Thus a relation between absorbed dose and exposure may be established in air or in other media.

D. Dose Equivalent

Research has shown that the effect of ionizing radiations on biological systems is not related exclusively to absorbed dose. The effects observed depend on many factors and include:

1. Type of radiation
2. Energy of the radiation
3. Distribution and/or fractionation of the radiation
4. Biological endpoint chosen
5. Time at which the endpoint is evaluated
6. Species and strain of the organism utilized.

Many other factors, such as the administration of certain drugs, also influence the response to radiation. In an effort to provide a comparable numerical basis for the effects of various types of radiation for protection purposes, a quantity termed the "dose equivalent" (DE) has been defined as:

> The product of the absorbed dose (D), quality factor (QF), and other necessary modifying factors:

$$DE = D\ (QF)\ (\ .\ .\ .\ ).$$

Depending upon the effect chosen for evaluation, the result obtained when the effects of two types of radiation are compared (frequently termed in radiobiological research the "relative biological effectiveness" or RBE) may vary widely. A representative value for the RBEs is selected by a committee to be used for radiation protection purposes. This value is termed the "quality factor" (QF).

Typical quality factors, given here, are taken from Rees, D. J., Health Physics: Principles of Radiation Protection (London: Butterworth, 1967):

Quality Factor	Type of Radiation
1.0	X rays; γ rays; electrons; and β rays with E_{max} greater than 0.03 MeV
1.7	β rays with E_{max} not greater than 0.03 MeV
10.*	Neutrons and protons up to 10 MeV
10.	Naturally occurring α particles
20.	Heavy recoil nuclei

* 30. in the case of irradiation of the eyes.

E. Activity

A quantity is necessary to describe the rate at which a radionuclide undergoes nuclear transformations. Historically, radium was used as a standard, but ICRU recommendations have established a quantity which is independent of any particular substance and depends only on measurable quantities, i.e., the number of transformations occurring and the time during which these emissions took place. The quantity involved is "activity" and is defined as follows:

> The activity (A) of a quantity of a radioactive nuclide is the quotient of ΔN by Δt where ΔN is the number of nuclear transformations which occur in this quantity in time Δt.
>
> $$A = \frac{\Delta N}{\Delta t}.$$

Activity applies only to the number of nuclear transformations taking place per unit time and is not directly related to the exposure rate, absorbed dose rate, or dose equivalent rate due to the radionuclide.

F. Units

Each of the quantities discussed has a measure or "unit" assigned to it. Exposure is expressed in terms of the equation

$$X = \frac{\Delta Q}{\Delta m},$$

where Q represents electrical charge (produced by ionization) per unit mass of air. Hence, any units of charge over mass would conform with the definition. The ICRU has, however, established a special unit, the use of which is restricted to measurement of exposure only. The unit of charge used is the "coulomb," and the unit mass is the "kilogram." Specifically,

$$1 \text{ roentgen} = 2.58 \times 10^{-4} \text{ coulombs/kilogram*}$$

Absorbed dose, defined in terms of energy deposited in a mass, would demand units of energy (joules, for example) per unit mass (kilograms). Again, a special unit has been defined which is used in referring to absorbed dose:

$$1 \text{ rad} = {}^{1}/_{100} \text{ joule/kilogram}$$
$$= 100 \text{ ergs/gram.}$$

The unit of dose equivalent is the "rem," which is obtained by multiplying the dose in rads by the appropriate quality factor and other modifying factors. The ICRU notes in its report that this statement ". . . does not cover a number of theoretical aspects (in particular the physical dimensions of some of the quantities) . . . " but that " . . . it fulfills the immediate requirement for an unequivocal specification of a scale that may be used for numerical expression in radiation protection." In short, since dimensionality is ignored, rems = rads × QF. Since the QF for x and gamma radiation is 1, the number of rems is identical to the absorbed dose in rads for these radiations.

The special unit for activity is the "curie." The commission has recommended that steps be taken to redefine the curie as

$$1 \text{ curie} = 3.7 \times 10^{10} \text{ s}^{-1} \text{ (exactly).}$$

The earlier definition of the curie was based on the activity of the radon gas in equilibrium with one gram of radium. Recent measurements have shown that the activity associated with this standard is $3.61 \times 10^{10} \text{ s}^{-1}$ and hence the unit has been redefined as a number independent of any given radioactive nuclide.

III. PRACTICAL CONSIDERATIONS

In the application of instrumentation to the assessment of the radiation fields, care must be taken to select instruments which will truly indicate or measure the radiation quantity of interest. Exposure can only be measured directly by an air-filled device which measures ionization in air. In practice, most exposure measuring instruments are related to a free-air chamber, and do not actually measure exposure directly. Most of the ionization produced and measured originates in the chamber wall--which is hopefully an "air equivalent" material.

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*The magnitude of this unit does not differ from the magnitude of the previous definition.

IV. SUMMARY

Certain of the salient features of radiation quantities and units are presented in the following table:

Associated Features	Quantities		
	Exposure	Absorbed Dose	Dose Equivalent
Radiation types	x- or gamma ray	all ionizing	all ionizing
Media in which effects are measured	air	any	biological system
Effects observed	ionization	energy deposited	biological effect
Units	roentgen	rad	rem

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Units of Radioactive Decay and the Decay Law

I. INTRODUCTION

The most widely used unit of quantity of radioactive material is the curie.

II. THE CURIE UNIT

A. Original Definition

Originally, the curie unit applied only to radium. It was based on the disintegrations per second occurring in the quantity of radon gas in equilibrium with one gram of radium. If permitted to attain this equilibrium, one gram of radium will produce about 0.66 mm^3 of radon. In this quantity of radon, about 37 billion atoms disintegrate each second.

B. Later Modifications

The International Radium Standard Commission extended the definition in 1930 to include that quantity of any radioactive decay product of radium which underwent the same number of disintegrations per second as one gram of radium. It avoided specifying the figure exactly, so for some years the exact value of the curie unit varied with each successive refinement in the measurement of the decay constant or the atomic weight of radium.

C. Modern Practice

In 1950, the International Joint Commission on Standards, Units, and Constants or Radioactivity redefined the curie unit by accepting 37 billion disintegrations per second as a curie of radioactivity regardless of its source or characteristics. At present, the unit of quantity of radioactivity, the curie, merely requires that in the given sample of any material, 3.7×10^{10} disintegrations occur each second. Smaller--and often more convenient--units are the millicurie, (one-thousandth of a curie; mCi), and the microcurie, (one-millionth of a curie; μCi). The picocurie (3.7×10^{-2} dis/sec or 2.22 dis/min) is often used to express the very low natural and environmental levels.

D. Curies and Grams: Specific Activity

The term curie originated from the number of emanations from one gram of radium. Thus, the activity of one gram of radium is equivalent to one curie. It is important, however, to note that when applied to radionuclides other than radium, the curie unit does not make apparent what weight of the material is required. Since the curie of activity is 37 billion disintegrations per second, the weight of the material required to produce this number of disintegrations per second will be a function of the decay rate of the atoms of the material (i.e., the disintegration constant) and of the number of atoms of the material per gram (i.e., gram atomic weight). As examples, a curie of pure ^{60}Co would weigh less than 0.9 milligram, whereas a curie of natural ^{238}U would require over two metric tons of the metal. "Curies per gram" is termed "specific activity."

III. THE RADIOACTIVE DECAY LAW

A. The Disintegration Constant

The activity of any sample of radioactive material decreases or decays at a fixed rate which is a characteristic of that particular radionuclide. No known physical or chemical agents (such as temperature, pressure, dissolution, or combination) may be made to influence this rate. The rate may be characterized in at least two ways, one of which is the use of the disintegration constant (λ). The disintegration constant (λ) represents the fraction of the total number of atoms present which decays in a unit time. Thus the number of disintegrations occurring per unit time in a given sample is the product of the number of atoms present in the sample, (N), and the fraction of these disintegrating in each unit time, (λ), or:

$$-\frac{dN}{dt} = \lambda N$$

where the minus sign is used to indicate that the number of atoms is diminishing. Integration of the above equation leads to the basic law of radioactive decay:

$$N_t = N_0 e^{-\lambda t}$$

Written out, this equation reads: the number of atoms (N_t) remaining after a time (t) is equal to the original number (N_0) multiplied by $e^{-\lambda t}$, where (e) is the base of the natural system of logarithms and (λ) is the disintegration constant.

B. The Half-Life

The disintegration constant is not as convenient as another method of representing the rate of radioactive decay. This is the half-life, or $T_{1/2}$, of the radionuclide. It is the length of time required for one half of the original number of nuclei to disintegrate. The relationship of the half-life to the disintegration constant (λ) can be shown in the following way:

$$\lambda = \frac{0.693}{T_{1/2}}$$

Therefore, one may substitute this expression for (λ) in the basic decay law, and

$$N_t = N_0 e^{-\frac{0.693\,t}{T_{1/2}}}$$

If it is desired to have the equation in terms of activity instead of number of atoms, one can multiply both sides of the equation by the disintegration constant (λ) as follows:

$$\lambda N_t = N_0 e^{-\frac{0.693\,t}{T_{1/2}}}$$

but $\lambda N = A$

therefore:

$$A_t = A_0 e^{-\frac{0.693\,t}{T_{1/2}}}$$

This is the working equation for computing the amount of radionuclide which will remain from the original sample after an interval of time elapses.

C. Computational Method

Example:

Given: $A_0 = 10$ mCi of ^{32}P

$t = 120$ days

$T_{\frac{1}{2}} = 14.2$ days$^{(1)}$

Find A_t the quantity remaining after 120 days

$$A_t = (10)\, e^{-\frac{(0.693)(120)}{14.2}}$$

$$= 10\, e^{-5.85} = 10\,(0.00288)$$

$$A_t = 0.0288 \text{ mCi}$$

This is the sort of calculation which would be required before diluting a radioisotope for future use or for determining the remaining activity in a quantity of nuclide which had been stored for some time since its standardization.

D. Graphic Method

There is an easy graphical method of accomplishing this same computation. It is based on the relation that each successive half-life reduces the activity present by one-half, and the effect is cumulative: i.e., two half-lives reduce to $\frac{1}{2} \times \frac{1}{2}$ or $\frac{1}{4}$; three reduce to $\frac{1}{2} \times \frac{1}{2} \times \frac{1}{2}$ or $\frac{1}{8}$, etc. In the general case,

$$A_n = A_0(\tfrac{1}{2})^n$$

where (n) is the number of half-lives elapsed. This function is graphed in the Radiological Health Handbook under Radioactive Decay, Semi-Log Plot. The answer to the "example" problem may easily be read from the graph at the point where the line intersects the horizontal axis. This number (8.44) is the number of half-lives of ^{32}P in a 120-day period. A useful "rule of thumb" to remember is that seven half-lives will reduce any activity to below 1 percent of its original value.

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This material was prepared by Mr. C. D. Geilker, formerly of the Training Branch, Division of Radiological Health.

Series Decay

I. INTRODUCTION

Series decay concerns the mathematical relationship of quantities of activity present when two or more radionuclides exist in a decay chain. Examples of a decay chain are the "natural decay series" given in the *Radiological Health Handbook*(1) or a two-step fission product decay series such as:

$$^{90}\mathrm{Sr} \xrightarrow{\beta^-} {}^{90}\mathrm{Y} \xrightarrow{\beta^-} {}^{90}\mathrm{Zr} \qquad \text{(stable)}$$

The relationship between three or more radionuclides is described by H. Bateman.(2) The solution, while straightforward, is quite involved. A two-step relationship (parent-daughter) can readily be derived and is reasonably easy to work with.

II. PARENT-DAUGHTER RELATIONSHIPS

A. General Relationships(3,4)

Decay of the parent nuclide is described by the equation:

$$\frac{dN_1}{dt} = -N_1\lambda_1$$

or

$$dN_1 = -N_1\lambda_1 \, dt$$

For the daughter nuclide:

$$dN_2 = N_1\lambda_1 \, dt - N_2\lambda_2 \, dt$$

where $N_1\lambda_1$ represents the rate at which atoms of the daughter are formed, and $N_2\lambda_2$ represents their rate of disappearance.

Upon integration,

$$N_2 = \frac{\lambda_1}{\lambda_2 - \lambda_1} N_1^0 \left(e^{-\lambda_1 t} - e^{-\lambda_2 t}\right) + N_2^0 e^{-\lambda_2 t}$$

where:

N_1^0 = initial number of parent atoms present

N_2^0 = initial number of daughter atoms present

N_2 - number of daughter atoms at time t

λ_1 = decay constant of parent

λ_2 - decay constant of daughter

The equation for N_2 can be converted to units of activity by multiplying both sides of the equation by λ_2

$$A_2 = \frac{\lambda_2}{\lambda_2 - \lambda_1} A_1^0 \left(e^{-\lambda_1 t} - e^{-\lambda_2 t}\right) + A_2^0 e^{-\lambda_2 t}$$

This equation is the most general form and can be used for any parent-daughter relationship. However, the equation may be simplified if relative magnitudes of λ_1 and λ_2 are specified. These special cases greatly simplify the mathematics involved.

B. Special Cases

1. Secular Equilibrium: $\lambda_2 >> \lambda_1$

In secular equilibrium the half-life of the parent is very much longer than the half-life of the daughter. Buildup of activity in a freshly-isolated parent fraction occurs as shown in Figure 1.

For the period up to equilibrium (about seven daughter half-lives), the general equation reduces to:

$$A_2 = A_1^0 \left(1 - e^{-\lambda_2 t}\right)$$

At secular equilibrium $A_2 = A_1$, i.e., the activity of the daughter is equal to the activity of the parent.

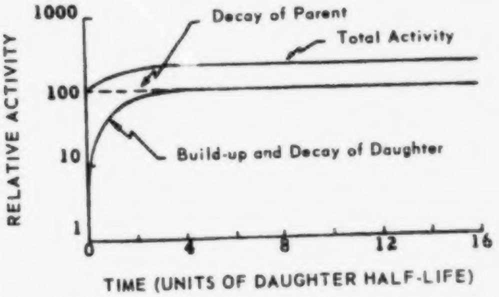

Figure 1.--Parent-Daughter Relationship in Secular Equilibrium

After significant decay of the parent

$$A_2 = A_1^0 e^{-\lambda_1 t}$$

Example: Decay of ^{90}Sr + ^{90}Y

2. Transient Equilibrium: $\lambda_2 > \lambda_1$

In transient equilibrium the half-life of the parent is longer than that of the daughter, but not very long. In a freshly purified parent fraction, the daughter activity builds up, then decays with the same half-life as the parent. This relationship is illustrated in Figure 2.

At equilibrium:

$$A_2 = \frac{\lambda_2}{\lambda_2 - \lambda_1} A_1$$

Note that the daughter activity is greater than that of the parent by the factor

$$\frac{\lambda_2}{\lambda_2 - \lambda_1}$$

Example: Decay of ^{140}Ba + ^{140}La

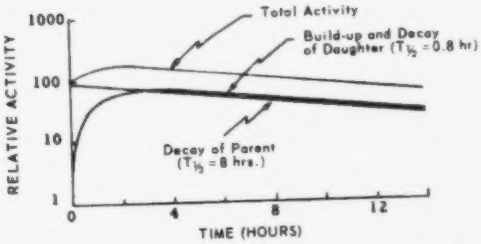

Figure 2.--Parent-Daughter Relationship in Transient Equilibrium

3. No Equilibrium $\lambda_1 > \lambda_2$

Here the half-life of the parent is shorter than that of the daughter. The relationship of the parent and daughter in this case is as shown in Figure 3.

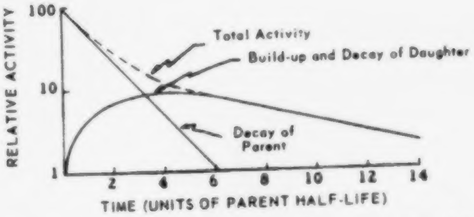

Figure 3.--Parent-Daughter Relationship Where No Equilibrium Exists

III. SERIES RELATIONSHIPS FOR NATURALLY OCCURRING RADIOACTIVE MATERIALS

A. General

Where three or more radionuclides exist in a decay series, as exemplified by the radioactive heavy elements occurring in nature, H. Bateman has formulated a relationship(1) for any member of such a chain.

From the radiological health standpoint, the uranium series and particularly radium and daughter product decay are most significant.

B. Examples

1. ^{238}U and ^{226}Ra Relationship (Secular Equilibrium)

At equilibrium: $A_1 = A_2$

where:

A_1 = activity of ^{238}U

A_2 = activity of ^{226}Ra

or, $\lambda_1 N_1 = \lambda_2 N_2$

or, $\frac{N_1}{N_2} = \frac{T_1}{T_2} = \frac{4.51 \times 10^9 \text{ years}}{1620 \text{ years}} = 2.8 \times 10^6$

where:

N_1 = number of atoms of ^{238}U

N_2 = number of atoms of ^{226}Ra

T_1 = half-life of ^{238}U

T_2 = half-life of ^{226}Ra

In natural uranium, therefore, one part of radium is present to about every three million parts of uranium.

2. ^{226}Ra and ^{222}Rn Relationship

At equilibrium: $A_1 = A_2$

Starting with pure radium, it would take about seven half-lives of ^{222}Rn ($7 \times 3.8 = 27$ days) for 99% equilibrium to be established.

3. ^{222}Rn and Daughter Products Relationship(5)

The daughter products of ^{222}Rn (i.e., ^{218}Po, ^{214}Pb, ^{214}Bi and ^{214}Po) are short-lived compared to the parent. Starting with pure radon, equilibrium is established in about 4 hours, after which approximately equal amounts (curie-wise) of radon and daughters would be present.

Starting with pure ^{218}Po (Ra A) the relation of this nuclide and several of its daughter products is as shown in Figure 4. The series ends as stable lead.

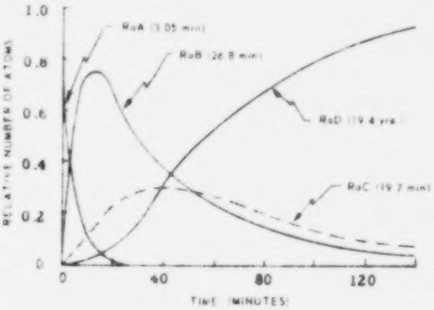

Figure 4.--The Decay of RaA and Daughter Products

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This material was prepared by Mr. C. D. Geilker and revised by Mr. L. J. Perez, both formerly of the Training Branch, Division of Radiological Health.

Interaction Of Radiation With Matter

I. INTRODUCTION

A. All radiation possesses energy either inherently, as in the case of electromagnetic radiation, or as kinetic energy of motion in the case of the particulate radiations. Absorption of radiation is the process of transferring this energy to atoms of the medium through which the radiation is passing. To say that radiation interacts with matter is to say that it is either scattered or absorbed. The mechanisms of the absorption of radiation are of fundamental interest in the field of radiological health primarily because of the following reasons:

1. Absorption in body tissues may result in physiological injury.

2. Absorption is the principle upon which detection is based.

3. The degree of absorption or type of interaction is a primary factor in determining shielding requirements.

B. The transfer of energy from the emitted particle or photon to atoms of the absorbing material may occur by several mechanisms but, of the radiations commonly encountered, the following two are the most important:

1. Ionization--any process which results in the removal of an electron (negative charge) from an atom or molecule thereby leaving the atom or molecule with a net positive charge.

2. Excitation--the addition of energy to an atomic or molecular system, thereby transferring it from its ground state to an excited state. Depending on the type of interaction, either the atomic nucleus or one of its orbital electrons may absorb the excitation energy.

C. Ordinarily, the atoms in a material are electrically neutral, i. e., they have exactly as many negative electrons in orbit as there are positive protons in the nucleus. Thus the difference, or net electrical charge, is zero. Radiations have the ability to either tear loose one or more of the electrons from their orbit or raise the orbital electrons to a higher energy level. In the former case (ionization) there is left behind an atom carrying an excess of positive charge, and a free electron. In the latter case (electronic excitation), the excited atom may lose its excess energy when an electron in a higher energy shell fails into the lower energy vacancy created in the excitation process. When this occurs, the excess energy is liberated as a photon of electromagnetic radiation (which may escape from the material but usually undergoes other absorptive processes.) Nuclear excitation is of significance only for neutrons or other radiation of relatively high energies. A discussion of the types of radiations and their interactions might conveniently be divided into four main categories:

1. Heavy, positively charged particles: alpha particles, protons, deuterons, tritons, and possibly mesons, would exhibit similar mechanisms of interaction with matter.

*Alpha particles will be discussed as a prototype of this group.

2. Beta particles: Both positions and electrons would fall into this category. They have equal masses and equal but opposite charges, and therefore lose their kinetic energy by similar mechanisms. The electron is used as the prototype of this category.

3. Electromagnetic radiation: This group includes both x- and gamma radiation; however since these two radiations differ in their origin and not in their mechanism of interaction, only gamma radiation will be discussed.

4. Neutrons: The interaction of neutrons is discussed separately in another section of this manual. Therefore, to avoid duplication, this radiation will not be described herein.

II. ALPHA ABSORPTION

A. An alpha particle is made up of two protons (positively charged) and two neutrons, all strongly bound together by nuclear forces. If such a particle approaches an electron (negatively charged), it experiences a strong electrostatic attraction, whereas if it approaches an atomic nucleus (also positively charged) it will tend to be repelled. Alpha particles have a mass about 8,000 times that of the electron. They are ejected from the nuclei of radioactive atoms with velocities of the order of 1/20 the speed of light. All of these properties--its large mass, its charge, and its high velocity tend to make the alpha particle an efficient projectile when it encounters atoms of an absorbing material. In other words, it would have a high probability of interacting, or colliding, with orbital electrons, and also atomic nuclei.

B. When speaking of "collisons" between subatomic particles, it should be understood that the particles (for example an alpha and an atomic nucleus) need approach each other only sufficiently close for forces to interact. Such an interaction may then be referred to as an collision. An alpha may undergo either elastic (no transfer of energy), or inelastic (transfer of energy from the alpha radiation to the target particle) collisions.

Inelastic collisions may result in ionization and/or excitation. And since a finite amount of energy is required to ionize or excite an atom, the kinetic energy of the alpha particle is gradually dissipated by such interactions until it captures two electrons and settles down to a quiet existance as a helium atom.

C. Due to the high probability of interaction between an alpha particle and orbital electrons of the absorbing medium, a large number of ion pairs are formed per unit path length. And since a fraction of the kinetic energy of the alpha particle is absorbed on formation of each ion pair, this type of radiation loses its energy over a relatively short distance. For these reasons, the range of alpha particles is much less than the range of other forms of radiation. It is, in summary a highly ionizing, weakly penetrating radiation.

1. Alpha particles from a given radionuclide are all emitted with the same energy, consequently those emitted from a given source will have approximately the same range in a material. Alpha particle range is usually expressed in centimeters of air.

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*Due to their large mass, heavier particles, such as fission fragments, would lose energy predominantly by other processes.

The relationship between range and energy has been expressed empirically as follows:

$$R_a = 0.318\ E^{3/2}$$

Where:

R_a = range in cm. of air at 1 atm. and 15° C

E = energy in Mev

The range of alpha radiation in solids may be compared to their range in air. The approximate relationship between range in a solid and range in air is given below:

$$R_{solid} = \frac{3.2 \quad 10^{-4} R_{air} \sqrt{A_{solid}}}{\rho_{solid}}$$

R = range

ρ = density

A = mass number

2. The number of ion pairs formed per centimeter of path in any given medium is called the specific ionization for that particular ionizing radiation.

$$\text{Specific ionization} = \frac{\text{number of ion pairs formed}}{\text{Centimeters of path}}$$

3. About 34 electron volts of energy are required for the formation of a primary ion pair in air. Only about half to two-thirds of this is required to remove the orbital electron, the balance being lost in electronic excitation processes. Depending on the energy of the alpha particle, the number of ion pairs formed per centimeter of path in air will range from 5,000 to 80,000.

III. BETA ABSORPTION

The rest mass of a beta particle is the same as that of an orbital electron and, as a consequence, very much smaller than the mass of the nuclei of the atoms making up the absorbing medium. Since beta particles and orbital electrons have like charges, they experience an electrostatic repulsion when in the vicinity of one another. But a beta particle has a charge opposite to that on the atomic nucleus, therefore an electrostatic attraction will be experienced as the beta approaches the nucleus.

All of these factors must be realized in order to understand the interaction between the beta radiation and atoms in an absorbing medium. For example, considering only the mass relationship between the beta particles and orbital electrons, one might expect that the interaction between these two electrons is somewhat similar to the collisions between billiard balls.

Actually, a beta particle may lose all of its energy in a single collision. In such an interaction, the target electron can effectively become an ionizing particle.

Normally a beta particle loses its energy in a large number of ionization and excitation events in a manner analogous to the alpha particle. Due to the smaller size and charge of the electron, however, there is a lower probability of beta radiation interacting in a given medium; consequently, the range of a beta is considerably greater than an alpha of comparable energy.

Since the electronic mass is small compared with that of a nucleus, large deflections can occur in single collisions, particularly when electrons of low energies are scattered by high atomic number elements (high positive charge on the nucleus). As a result, a beta usually travels a tortuous, winding path in an absorbing medium.

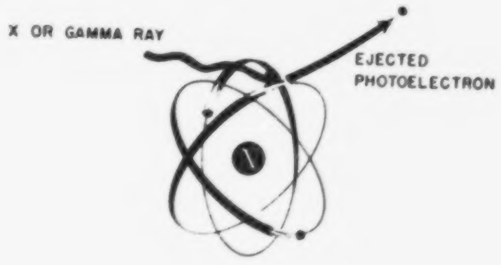
In addition to ionization and excitation, a beta may have an interaction with an atom which results in the production of x-rays. A high energy beta may penetrate through the electron cloud surrounding the nucleus of the atom, and in traveling through the various quantum energy states of the orbital electrons, it experiences the strong electrostatic force of the nucleus, resulting in a change in velocity and the emission of several x-rays having a spectrum of energies. (Such x-rays are referred to as "Bremsstrahlung radiation"). It becomes an increasingly important mechanism of energy loss as the initial energy of the beta increases, and the atomic number of the absorbing medium increases. As previously pointed out, all alpha particles from a given radionuclide are emitted with the same energy. When betas are emitted, the total kinetic energy involved in the decay of the radioactive atom is divided between the beta and neutrino. The neutrino has a zero charge and negligible mass, consequently its contribution to the overall interaction is insignificant, but it does carry away a fraction of the total kinetic energy available, therefore the beta may be emitted with an energy varying from practically zero up to a maximum energy (which is a characteristic of the particular radionuclide).

IV. GAMMA ABSORPTION

X- and gamma rays differ only in their origin, and an individual x-ray could not be distinguished from an individual gamma ray. Both are electromagnetic waves, or quanta, and differ from radio waves and visible light waves only in having much shorter wavelengths. The difference in name is used to indicate a difference in source--gamma rays are of nuclear origin, while x-rays are of extra-nuclear origin (i.e., they originate in the electron cloud surrounding the nucleus). Both x-rays and gamma rays have zero rest mass, no net electrical charge, and travel with the speed of light. They are basically only distortions in the electromagnetic field of space, and because of this fact, they interact electrically with atoms to produce ionization even though they themselves possess no net electrical charge. As previously pointed out, gamma rays will be discussed as the prototype of this type of radiation. There are three mechanisms by which gamma rays lose energy by interacting with matter.

A. The Photoelectric Effect

The photoelectric effect is an all-or-none energy loss. The gamma wave, or photon, imparts all of its energy to an orbital electron of some atom. The gamma photon, since it consisted only of energy in the first place, simply vanishes. The energy is imparted to the orbital electron in the form of kinetic energy of motion, and this greatly increased energy overcomes the attractive force of the nucleus for the electron and causes the electron to fly from its orbit with considerable velocity. Thus, an ion-pair results. The high velocity electron (which is called a photoelectron) has sufficient energy to knock other electrons from the orbits of other atoms, and it goes on its way producing secondary ion-pairs until all of its energy is expended.

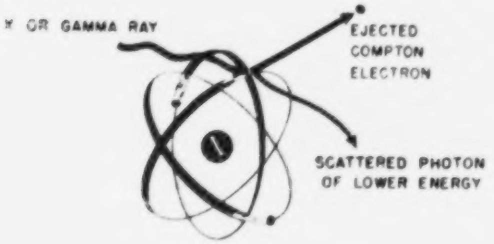


PHOTOELECTRIC EFFECT
(PRIMARILY LOW ENERGY PHOTON)

Figure 1a

B. The Compton Effect

The Compton Effect provides a means of partial energy loss for the incoming gamma ray. Again the gamma ray appears to interact with an orbital electron of some atom, but in the case of Compton interactions, only a part of the energy is transferred to the electron, and the gamma ray "staggers on" in a weakened condition. The high velocity electron, now referred to as a Compton electron, produces secondary ionization in the same manner as does the photoelectron, and the weakened gamma ray continues on until it loses more energy in another Compton interaction or disappears completely via the photoelectric effect. The unfortunate aspect of Compton interaction is that the direction of flight of the weakened gamma ray is frequently referred to as a "scattered" photon, and the entire process known as "Compton scattering". By this mechanism of interaction, the direction of photons in a beam may be randomized, so that scattered radiation may appear around corners and behind "shadow" type shields.



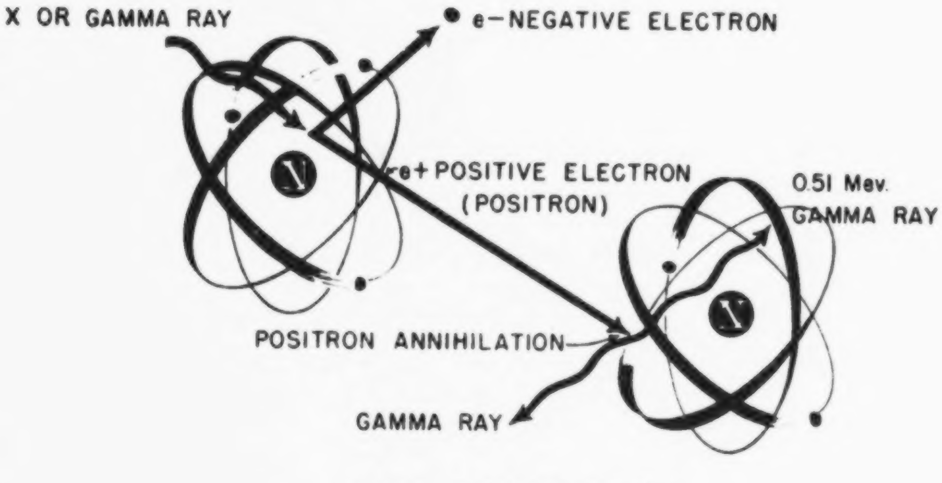
COMPTON EFFECT
(PRIMARILY MEDIUM ENERGY PHOTON)

Figure 1b

C. Pair Production

Pair production, the third type of interaction, is much rarer than either the photoelectric or Compton effect. In fact, pair production is impossible unless the gamma ray possesses at least 1.02 Mev of energy. (Practically speaking, it does not become important until two Mev of energy). In pair production a gamma photon simply disappears in the vicinity of a nucleus, and in its place appears a pair of electrons--one negative, one positive. The mass of these electrons has been created from the pure energy of the photon, according to the familiar Einstein equation $E = mc^2$, where (E) is energy, in ergs, (m) is mass in grams, and (c) is the velocity of light in cm/sec. If there is any excess energy in the photon above the one Mev required to create two electron masses, it is simply shared between the two electrons as kinetic energy of motion, and they fly out of the atom with great velocity. The negative electron behaves in exactly the ordinary way, producing secondary ion-pairs until it loses all of its energy of motion. The positive electron (known as a positron) also produces secondary ionization so long as it is in motion, but when it has lost its energy and slowed almost to a stop, it encounters a free (negative) electron somewhere in the material. The two are attracted by their opposite charges, and upon contact, annihilate each other, converting the mass of each into pure energy. Thus two gammas of 0.51 Mev arise at the site of the annihilation. The ultimate fate of the annihilation gammas is either photoelectric absorption or Compton scattering followed by photoelectric absorption.

Figure 1 illustrates the three mechanisms of gamma interaction, and Figure 2 shows the fraction of the total decrease in beam intensity which is due to the action of each, for energies between 10 kev and 100 Mev, using lead as the absorbing material. At low energies the photoelectric interactions predominate, at intermediate energies Compton scattering is the most likely interaction, and at high energies pair production is the most important.


X OR GAMMA RAY
e−NEGATIVE ELECTRON
e+ POSITIVE ELECTRON
(POSITRON)
0.51 Mev.
GAMMA RAY
N
N
POSITRON ANNIHILATION
GAMMA RAY

PAIR PRODUCTION
(HIGH ENERGY PHOTON, > 1.02 Mev.)

Figure 1c

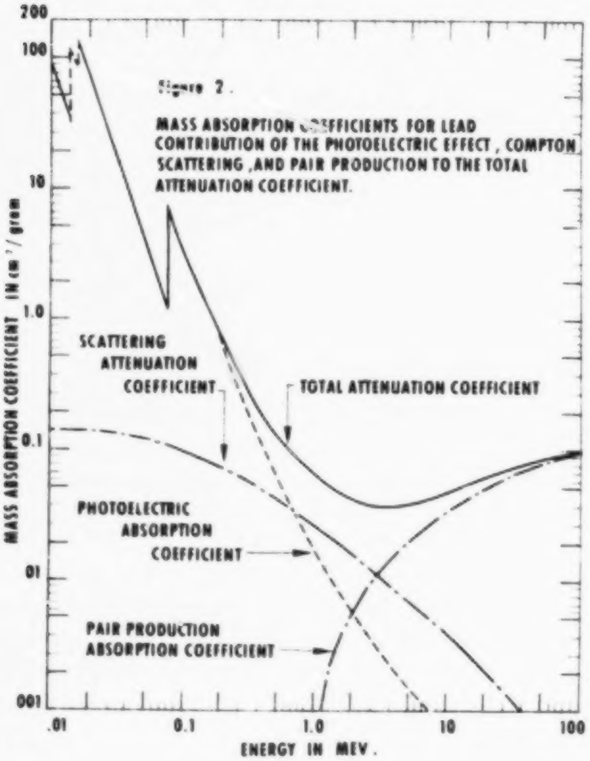


Figure 2

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GPO 876-255

Basic Principles of Radiation Detection Instruments

I. INTRODUCTION

Since man can neither sense nor measure the presence of radiation, instruments must be used for its detection. Such instruments measure or respond to the charged particles which are produced as radiation interacts with matter. The basic difference between the various instruments is the medium in which the ionizing events occur.

The principal means of detection are: gas ionization, photographic emulsions, scintillation media, semiconductors, chemical decomposition media, radiophotoluminescence and optical density, thermoluminescence, and calorimetry. They will be discussed in the paragraphs that follow.

II. GAS IONIZATION

Radiation detection devices, based on the principle of collecting ions formed by the interaction of ionizing radiation in the chamber wall and enclosed gas, comprise a large segment of all radiation detection instruments now in use. The gases usually used are air or tissue-equivalent gas mixtures, or a mixture of a noble gas and a small amount of a polyatomic gas such as methane or isobutane. The content, concentration, and pressure will vary with the specific function of the instrument.

A. Regions of Response

The relationship of the several types of gas ionization instruments can best be illustrated by a hypothetical experiment which utilizes a detection chamber, a variable voltage supply, and a current indicator of high sensitivity and wide range.

In this experiment, if we expose the detection chamber to a constant source of radiation and apply an increasing voltage across the chamber, the measured current output produced by ionization within the chamber will yield five regions of response, as illustrated in Figure 1.

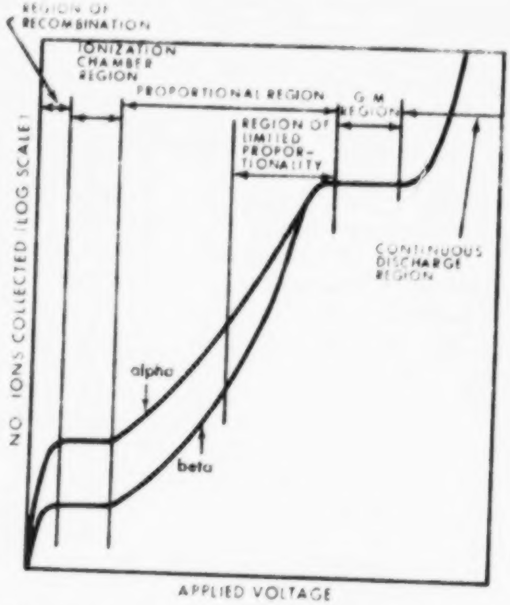

Figure 1.--Regions of Instrument Response

1. RECOMBINATION REGION

In the first region, the ions produced by the radiation will be under very low voltage gradients and will tend to recombine with each other rather than migrate to the electrodes and be collected. This recombination of ion pairs decreases as the applied voltage is increased and finally becomes negligible. At some voltage the field strength will be sufficient to collect essentially all of the ion pairs that are formed. This first region is known as the region of recombination and is usually not useful for the operation of radiation detection instruments.

2. SATURATION OR IONIZATION CHAMBER REGION

This region begins at the voltage at which all ions formed are collected. These are the primary ions resulting from the action of the radiation, and are comprised of ion pairs. The negative portion of the ion pair (electron) is accelerated toward the anode or positive electrode of the chamber, while the positive ion (residue of the atom) is drawn more slowly toward the cathode or negative electrode. For some large increment of voltage above the region of recombination, all ions produced in the gas by radiation are collected by the electrodes. As the voltage is increased within this region, the ions are given more energy and move faster toward the electrodes. However, they do not become energetic enough to produce additional ionization. This second region provides the first of the three operating regions for gas ionization instruments.

3. PROPORTIONAL REGION

If the voltage is increased above the ionization chamber region, the number of ions collected by the electrodes is greater than the number produced by the radiation. Under the voltage gradient, the primary electrons achieve a high enough velocity to cause secondary ionization in the filling gas. This secondary ionization results in an amplification of the number of ion pairs produced by the radiation. Hence, each primary ion pair produces several additional ions which are collected and measured on the instrument.

The number of secondary ions produced for each primary ion pair formed by the radiation is called the "gas amplification factor." As the voltage is raised, the gas amplification factor is increased. Gas amplification factors are normally about 10^3; factors as high as 10^5 or 10^6 are sometimes attained in this region. The number of ions collected is related to the applied voltage, and is proportional to the number of primary ion pairs formed. Thus, an alpha particle with its high specific ionization will produce a much larger pulse of current than will beta radiation with its correspondingly lower energy and lower specific ionization, illustrated by the alpha and beta curves respectively in Figure 1. This fact makes possible pulse height discrimination between alpha and beta which differ in the amount of primary ionization produced.

This proportionality (of the number of ions collected to the primary ionization) is the reason for naming the third region of instrument operation "the proportional region." The upper part of this region, where the two curves begin to approach each other (see Figure 1), is referred to as the "region of limited proportionality" and is not generally used in radiation instrumentation. This limit is controlled by the physical dimensions of the counter and the number of gas atoms present. The true proportional relationship no longer exists in this upper region.

4. GEIGER-MUELLER REGION

In the G-M region the increased voltage accelerates the primary electrons. These latter interact with gas molecules, producing a sequence of ionizing events during their travel to the anode. When the secondary ionization does start, it builds up rapidly, since the secondary ionizing events can also produce more ionization. The buildup of ionization, referred to as a "Townsend avalanche," then collects on the central electrode.

Since a single ionizing event can produce a very large number of ions, the number of ions collected is relatively independent of the applied voltage and the specific ionization of the incident radiation. G-M counters operating in the G-M region provide a gas amplification factor as high as 10^{10} and are extremely sensitive to any radiation which produces even one ion pair. Consequently, individual ionizing events may be detected.

The positive ions (often argon) produced during this avalanche of electrons migrate to the cathode, where their electron deficiency is satisfied by the excess negative charge at the cathode. When the vacant orbits of the gas atoms are filled, electromagnetic radiation is emitted. This radiation, which may be either ultraviolet or x radiation, tends to continue the discharge action, and so sustain the period during which the gas is essentially a conducting medium.

5. CONTINUOUS DISCHARGE REGION

If the voltage is increased above the G-M region the gas arcs, thereby producing a state of continuous discharge.

The experiment described (utilizing a detection chamber, a variable voltage supply, and a current indicator of high sensitivity and wide range) is theoretically possible. But in practice one does not convert an ionization chamber to a proportional counter or G-M counter by merely raising the voltage. For practical reasons these instrument types differ from each other not only in voltage but in configuration and composition of the contained gas.

B. Operational and Practical Considerations

The operational characteristics of detection instruments occur primarily in three of the five regions of instrument response. The first--the ionization chamber region--provides low sensitivity but high range, since it measures only the primary ionization produced. Discrimination between the several types of radiation is not possible except by use of external absorbers. The operating voltage for ionization chamber instruments will usually be between 60 and 600 volts, depending upon the filling gas (usually air at atmospheric pressure) and the physical size and shape of the chamber.

Proportional instruments provide a high sensitivity due to their gas amplification and a (relatively) high range, since the secondary ionization takes place over only a portion of the chamber volume. Due to the proportionality factor which exists in this region, the instrument is inherently capable of discriminating between different types of radiation. Proportional counters are usually filled with a mixture of argon and methane, although air is sometimes used. Operating voltages range from 500 to 5000 volts, depending upon chamber design and filling gas.

The G-M region provides high sensitivity, since any ionizing event occurring within the G-M tube can be counted. It has a low range due to the discharge dead time (the time during which the gas is conducting and therefore insensitive to any further ionizing events). Because of the nature of the discharge, it is impossible to discriminate between the various types of radiation in this region. Geiger-Mueller chambers are usually filled with argon or helium (and in most cases, a quenching vapor) at lower than atmospheric pressure. They operate in the range of 600 to 3000 volts.

To suppress the secondary emission of electrons from the cathode, a quenching agent is used. Organic quenching agents, usually polyatomic molecules such as ethyl alcohol, absorb the energy from the positively charged ion and dissociate into smaller particles which do not emit ultraviolet light and hence cannot eject electrons from the cathode. The useful life of the organically quenched tube is limited by the number of quenching molecules present. Organically quenched tubes have a reasonably flat plateau. The use of halogen gases will produce the same quenching effect; however, the halogen ions apparently recombine after dissociation and therefore do not limit the life of the tube.

The G-M region is illustrated in Figure 2. The threshold voltage is that applied voltage where the pulses are first detected and the avalanche effect results with further voltage increase. Normal operating voltage is selected at approximately one-third the plateau distance to obtain the greatest stability and life of the tube. Sustained operation in the continuous discharge region will result in permanent damage to the G-M tube.

Each of the regions provides certain operating characteristics described above

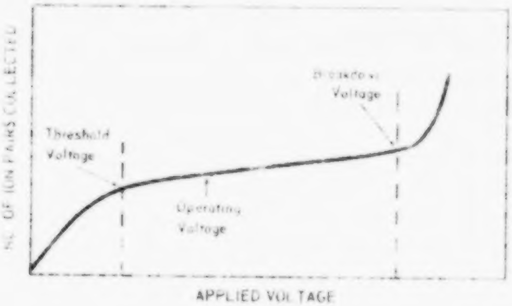

Figure 2.--G-M Response Curve

which makes it useful for one purpose or another: The ionization chamber region, providing a direct indication of the number of ions produced by a given radiation, is eminently suitable for indicating cumulative exposure or radiation exposure rate. Ion chambers are not sensitive to low radiation intensities, but complementary to their low sensitivity is their ability to measure large doses or high radiation intensities. Proportional instruments find their best use in the discrimination between alpha and beta radiation, while G-M instruments are extremely sensitive indicating devices for measuring low intensities of radiation. Both the proportional and the G-M instruments are counters; that is, they provide a pulse of current for every particle or photon which interacts within the chamber. These two types are useful in the measurement of radioactivity.

III. PHOTOGRAPHIC EMULSIONS

The first method used to detect nuclear radiation was the exposure of photographic plates. Becquerel thus discovered radioactivity. Radiation interaction with the silver halide in the photographic emulsion results in ionization. The silver ions produced are attracted to the negatively charged sensitivity center in the crystal, where the silver ions are reduced to free silver. This concentration of silver ions is known as the "latent image" and is proportional to the incident ionizing radiation. The latent image acts as a catalyst during the developing process to convert the grain to free silver. This results in an image amplification factor of approximately 10^{10}. The fixing process dissolves the non-reduced silver halide in the emulsion, leaving the final radiograph.

Photographic emulsions are used extensively in research, autoradiography, quality control, and personnel monitoring.

With the proper selection of film, filters, etc., the film exposure may be related to the type, energy, and quantity of radiation received by the film.

IV. SCINTILLATION MEDIA

Scintillation detectors were in use as early as 1908, and played an important part in the experiments of Rutherford and his collaborators. However, scintillation counting was not widely used until 1947, when the newly developed photomultiplier tube eliminated the tedium of counting scintillations under the microscope. The theory of scintillators is based on the luminescent property of certain materials. The interaction of radiation in a scintillation media results in absorption of energy. This energy raises a molecule or an ion to an excited or elevated energy state. The return to lower energy or metastable state levels, and finally decay to the ground state, results in the emission of energy as visible or near-visible light. The magnitude of each light pulse is proportional to the energy deposited in the medium. The energy deposited in the medium more closely approaches the total energy of the incident radiation as the volume of the scintillating medium increases. The combination of the phototube and suitable electronic circuits has resulted in a very effective counting system.

The sequence of transforming the energy of ionizing radiation into measurable electrical signals is illustrated in Figure 3. The advantages of the scintillation detector are the high efficiency in gamma ray detection compared to gas flow detectors, the capacity

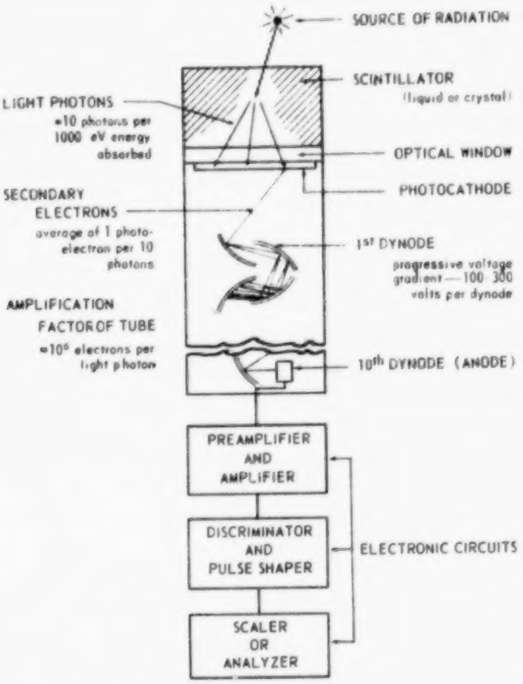
SOURCE OF RADIATION
SCINTILLATOR
(liquid or crystal)
LIGHT PHOTONS
$\approx$10 photons per 1000 eV energy absorbed
OPTICAL WINDOW
PHOTOCATHODE
SECONDARY ELECTRONS
average of 1 photo-electron per 10 photons
1st DYNODE
progressive voltage gradient — 100-300 volts per dynode
AMPLIFICATION FACTOR OF TUBE
$\approx 10^6$ electrons per light photon
10th DYNODE (ANODE)
PREAMPLIFIER AND AMPLIFIER
DISCRIMINATOR AND PULSE SHAPER
SCALER OR ANALYZER
ELECTRONIC CIRCUITS

Figure 3.--Schematic Diagram of Scintillator-Photomultiplier Counting System

to handle high counting rates because of the very short resolving time, and the ability to detect different types and energies of radiation. Perhaps the greatest use of scintillation counters is in measuring the energy spectrum of gamma emitters. Liquid scintillation systems have the advantage of high sensitivity, accuracy, reasonable stability and reproducible geometry. They have the disadvantage of poor resolution. Solid scintillation systems have high sensitivity and high resolution, but reproducible geometry is more difficult to attain.

Some of the most common scintillation media are: silver-activated zinc sulfide for the detection of alpha radiation, anthracene, naphthalene, stilbene, and liquid scintillators such as 2, 5-diphenyloxazole (PPO) and 2, 2-p-phenylenebis (5-phenyloxazole) (POPOP) for measuring beta radiation; and thallium-activated sodium iodide crystals for gamma detection.

V. SEMICONDUCTORS

Semiconductors use a dense ionizing medium, so photons of higher energy can be stopped completely within the medium. The ionizing events produce an electric field at the junction surface of two semiconductor materials. The most widely used types of semiconductor devices are diffused p-n junction, surface barrier, and lithium drift detectors.

A. Diffused p-n Junction

The diffused p-n junction detector obtains its name from its manufacturing process. A slice of p-type silicon or germanium crystal with a layer of n-type impurity (usually phosphorus) deposited on the surface is heated to form a p-n junction just below the surface. The phosphorus may also be painted onto the silicon and made to diffuse into it by applying heat. Since the n-type crystal has an excess of electrons and the p-type has an excess of "holes" (holes may be thought of as unit positive charges), the natural action of the crystal tends to align the electrons on one side of the junction and the holes on the other. Thus, a difference of potential is built up across the junction.

By applying an external voltage to the crystal of such polarity as to oppose the natural movement of electrons and holes (reverse bias) the potential barrier across the junction is increased and a "depletion region" produced. (See Figure 4.) This

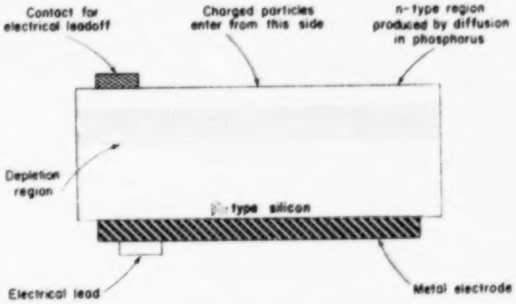
Contact for electrical leadoff
Charged particles enter from this side
n-type region produced by diffusion in phosphorus
Depletion region
p-type silicon
Electrical lead
Metal electrode

Figure 4.--Schematic Representative of a Diffused p-n Junction Detector

depletion region is the sensitive volume of the detector, and is analogous to the gas volume in a gas ionization detector. Charged particles entering the depletion region produce electron-hole pairs analogous to ion pairs produced in gas ionization chambers. Since an electric field exists in this region, the charge produced by the ionizing particle is collected, producing a pulse of current. The size of the pulse is proportional to the energy expended by the particle.

B. Surface Barrier Detectors

The operating principle of the surface barrier and lithium drift detectors is the same as for the p-n junction, in that a depletion region exists when an electric field is produced. The method of producing the depletion region, as well as its dimensions and location within the crystal, vary from one type to another.

The surface barrier detector depends for its operation upon the surface states of the silicon or germanium. At the surface of a piece of pure crystal, an electric field exists such that both holes and electrons are excluded from a thin region near the surface. For n-type crystals, the field is such as to repel the free electrons from this region. If a metal is joined to the crystal the free electrons are still repelled, but a concentration of holes is produced directly under the surface. If a reverse bias is then applied, a depletion region is produced. (See Figure 5.)

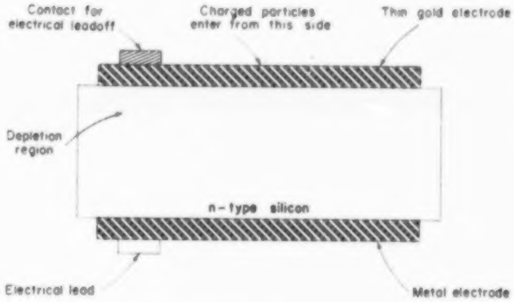

Figure 5.--Schematic Representation of a Surface Barrier Detector

Surface barrier detectors give better resolutions for particle spectroscopy than p-n junctions, but deeper depletion regions are possible with the latter. (The deeper the depletion region, the higher the energy of particles which can be analyzed, since the particle must expend all its energy in the depletion region.)

C. Lithium Drift Detectors

The lithium drift detector is produced by diffusing lithium into low resistivity p-type silicon. When heated to about 150° C under reverse bias, the lithium ions drift into the junction in such a way that the impurities are compensated to form a depletion region of high resistivity intrinsic silicon. Wide depletion regions may be obtained at low bias voltages in this manner.

Semiconductor detectors have been popular mostly in the area of charged particle spectroscopy. Much better resolution can be obtained in these devices than with scintillators or grid-type ionization chambers.

VI. CHEMICAL DECOMPOSITION INDICATORS

Since radiation can cause ionization, it is possible to use the ionization produced in a chemical system as an indication of the amount of radiation received. This is done in the case of chemical decomposition indicators: Ions produced by radiation combine chemically to form new compounds or change chemical characteristics from those existing in the pre-irradiation stage.

A typical chemical decomposition indicator is a chloroform-water mixture. When exposed to radiation, it produces hydrochloric acid in proportion to the radiation absorbed. This formation of acid decreases the pH. By the use of a suitable indicator, it is possible to ascertain when a predetermined dose has been received by the chemical system. An indicator frequently used in this system is bromcresol-purple.

An inherent drawback of chemical decomposition systems is the fact that the sensitivity to radiation is quite low. While these systems require higher exposure rates for detectable chemical change, newer systems are being developed to respond to lower exposures.

VII. RADIOPHOTOLUMINESCENCE AND OPTICAL ABSORPTION

Radiophotoluminescence is the phenomenon by which certain materials undergo changes in their photoluminescent properties after irradiation. Irradiated material will fluoresce when activated by light of the proper wavelength (ultraviolet or near-ultraviolet), while unirradiated glass will not fluoresce under the same conditions.

Silver-activated phosphate glass has proved a useful radiation detecting medium, exhibiting the photoluminescence property. The ionizing radiation liberates electrons within the glass. These are trapped by the Ag^+ ions of the glass. The resulting metallic silver centers serve as the origin of the photoluminescence. After radiation exposure of the glass, it is subjected to ultraviolet light and the resulting fluorescence detected by a photomultiplier tube. The intensity of the light emitted is proportional to the dose. Personnel dosimeters of this type have been developed which cover the range from 10 mR to 10^4 R.

Higher radiation doses may be determined by measuring the increased optical density of the glass after irradiation. This is done by making optical transmission measurements using light of the proper wavelength for the dose range of interest. This principle is useful from 10^3 to 10^6 rads.

VIII. THERMOLUMINESCENCE

Thermoluminescence is the phenomenon whereby certain materials can absorb and store energy from ionizing radiation, and release this energy as light in the visible or near-visible region of the spectrum when the material is heated. Irradiation of a luminescent material results in the excitation of electrons (see Figure 6) from the ground state to the conduction band of the material (a). The electrons are then trapped in a metastable state, present as imperfections in the crystal lattice of the material (b).

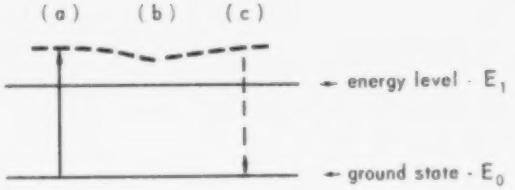

Figure 6.--Schematic of Electron Energy Bands

The electrons stored in the metastable state are activated by the addition of energy in the form of heat. The electron is then able to return to the ground state, and the energy released as light (c). The light intensity is proportional to the energy absorbed in the material from the original ionizing radiation. The TLD (thermoluminescent dosimetry) system requires a separate readout system which can measure either the integrated light output during, or the peak light output after, a constant heating cycle.

TLD is most widely used in personnel dosimetry and special research projects.

IX. CALORIMETRY

Calorimeters take advantage of radiation's heating effect to detect its presence. A calorimeter is a device for the measurement of quantities of heat. Calorimeters have received greatest application in the determination of the radiation-energy absorption in material, since ultimately the absorbed energy is degraded into heat. Calorimetry can also be used to determine the activity of a sample of radioactive material. The main advantage of the calorimetric method for measuring absorbed energy or activity is its inherent accuracy. For dosimetry purposes a direct reading of energy absorption can be obtained. However, the rate of heat input is so small that only very high intensities of radiation can be measured. For this reason, calorimetry is not used for routine monitoring purposes. Applications include the measurement of the activity of curie amounts of alpha emitters, the measurement of the energy of particles produced by particle accelerators, the dose from high intensity x-ray machines, etc. Calorimetry provides the only fundamental method for measuring absorbed dose.

X. SUMMARY

Radiation detection instrument design has been dictated by the exposure rate or activity to be measured, the type and energy of the radiation, accuracy desired, and the area of interest in which the data are to be used. Each of the primary means of detection discussed has inherent advantages and disadvantages. Laboratory and research studies should utilize all types of media for specific applications. Survey instruments use predominately gas ionization and scintillation media. Personnel monitoring instruments rely on gas ionization, photographic emulsions, and thermoluminescent media for detection of radiation.

As a result of these considerations, radiation detection instruments may be divided into three groups:

1. Personnel monitoring instruments, which measure accumulated exposure that can be related to the dose equivalent.
2. Survey instruments designed to measure exposure rate in milliroentgens per hour (mR/hr) or activity in counts per minute (cpm).
3. Laboratory instruments designed primarily to relate the ionizing events in terms of units of activity (curie).

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Survey Instruments

I. INTRODUCTION

None of the ionizing radiations is detectable by any of man's five senses, therefore all indications of their presence and intensity must be obtained by instruments. Radiation detection devices, like other measuring instruments, operate because of some effect the phenomenon being measured has on matter. In the case of radiation, this effect is ionization. Survey meters are similar to other radiation detection instruments in their operational characteristics. A good survey meter should be portable, rugged, sensitive, simple in construction, and reliable. Portability implies lightness and compactness with a suitable handle or strap for carrying. Ruggedness requires that an instrument be capable of withstanding mild shock without damage. Sensitivity demands an instrument which will respond to the type and energy of radiation being measured. Rarely does one find an instrument capable of measuring all types and energies of radiation that are encountered in practice. Simplicity in construction necessitates convenient arrangement of components and simple circuitry comprised of parts which may be replaced easily. Reliability is that attribute which implies ability to duplicate response under similar circumstances. All these conditions are not met in any one instrument, but they are approached in many. In any monitoring operation, one must select the proper instrument, use it intelligently, and then be able to interpret the results of the meter readings.

II. IONIZATION CHAMBERS

A. Theory

Ionization chambers are instruments in which the ionization initially produced within the chamber by radiation is measured without further gas-amplification. Primary ions formed in the chamber are attracted to the respective electrodes, and the current pulses are amplified externally to a measurable current. The gas-amplification factor is thus one.

B. Physical Description

Ionization chamber survey meters have three principal components: (1) the ionization chamber; (2) the electronic circuit; and (3) the dial or indicating meter.

(1) Ionization Chambers are usually about 30 to 50 cu. in. in volume and are filled with air at atmospheric pressure. The chamber wall design and type of material used in its construction determine the types of radiation to which it is sensitive. The larger the chamber the more sensitive the instrument and the greater the voltage required for proper operation. Practically all chambers have walls that conduct electricity and serve as the cathode, while wires mounted in the center of the chambers constitute the anode. Operating voltage is supplied by batteries and has a magnitude of about 100 volts. The current which flows is directly related to the type, energy, and quantity of radiation penetrating the chamber. With movable shields, as in the "Juno," it is possible to discriminate between types of

radiation. In general, ionization chamber survey meters are used to measure relatively high level intensities. Their low sensitivity enhances their capacity to measure radiation at higher dosages or exposure rates.

(2) The electronic circuit is actually a precision amplifier. Vacuum tube or transistor circuits are used to build up the feeble ionization current so that it may be measured directly by a microammeter. Most survey meters incorporate a system whereby the amplification characteristics of the circuit may be changed by factors of ten. This enables the operator to change the instruments' range and sensitivity.

Since there is a gas amplification factor of unity, circuit amplification becomes a problem which one does not necessarily have in a GM instrument. On the other hand, there is no problem of quenching the discharge or of losses due to coincidence.

(3) The indicating meter is usually a microammeter that registers the amplified current. The dial is generally calibrated in milliroentgens/hour or, in the case of contamination monitors such as the "Samson," in counts per minute.

C. Operation

Most ionization chamber survey instruments have a selector switch marked "off," "wait," and x1, x10, x100. When the switch is off, the batteries are disconnected and the meter is short-circuited making the instrument inoperative. With the switch in the wait position, the batteries are connected, permitting the circuit to warm up and the instrument to be zeroed after a warmup period of from 1 - 5 minutes. The meter is connected while the ionization chamber is disconnected making it possible to adjust the meter accurately to zero even in the presence of radiation.

The ionization chamber does not wear out or suffer changes in characteristics as GM tubes do; however, the circuit of the ionization chamber survey meter has more elements that can get out of adjustment if not properly handled. Loose leads and weak batteries are a source of trouble which can be readily serviced. Other difficulties are usually caused by faulty circuits which cannot generally be fixed without the aid of a competent service man.

No aural indication is used in IC instruments and thus the operator must constantly watch the meter to ascertain the field intensity. There is a lag between the instant radiation enters the chamber and the time when the meter reaches its maximum reading; therefore, one must allow time for the meter to reach its maximum before taking a reading. This is on the order of a few seconds.

D. Calibration

Instruments are designed by manufacturers to read directly in radiation intensity units, generally mr/hr or r/hr; however, there is considerable error in a direct reading, since the characteristics of individual components causes variations in instrument response. Each instrument must be calibrated for accurate interpretation. Instrument response intended by the manufacturer is related to one type of radiation, usually of a definite energy range. If radiation of a different energy or type is measured, the results will be incorrect and the instrument must be recalibrated with radiation of the same type and energy that is to be measured.

For gamma ray calibration an ionization chamber instrument can be checked by placing it in a known field of radiation. Radium and Co^{60} are the most frequently used sources for gamma calibration. A plot of scale readings versus true radiation intensity can be made by comparing scale readings at known distances from the source against the true intensities at these distances, employing the formulae given below.

$$I = \frac{\text{milligrams of Ra}}{D^2}, \text{ or } I$$

$$= \frac{1.59 \times \text{millicuries of Co}^{60}}{D^2}$$

where: I = Intensity, mr/hr

D = Distance from source to detector, yards

From five to ten equally spaced scale readings should be taken and a graph of these values versus the calculated intensities at the corresponding points should be plotted on linear graph paper. A typical calibration curve would plot meter readings on the ordinate and corresponding "true" intensities on the abscissa.

When calibrating an instrument, the reference point of the instrument is generally considered as the center of the sensitive volume. It should be pointed out that the ionization chamber type survey instrument, when properly calibrated, will give a good measurement of x- or gamma radiation intensity, but for alpha and beta radiation, only qualitative measurements can be made.

E. Uses

In x-ray survey work, calibrated ionization chamber instruments are very useful for measuring dose rate. Ion chambers are used extensively for beta and gamma survey work, and if properly modified, they may be used for neutron monitoring.

F. Typical Instruments

Cutie Pie: Perhaps the most widely used and one of the most versatile ionization chamber instruments available for radiological survey work is the Cutie Pie. These are available with maximum scale readings up to 50 roentgens/hour.

Condenser r-meters: A very reliable and accurate instrument for x-ray calibration is the condenser r-meter. By nature, the condenser r-meter measures cumulative dose. It consists of a charger-reader mechanism and several detachable ion-chambers. These chambers are charged and then left in a radiation field for a known time. When read on the charger-reader, they show the total dose received during the time of exposure. Ionization chambers for condenser r-meters are available in ranges from 0.025 roentgens full scale to 250 roentgens full scale and are nearly energy independent (- 2%) for x-ray energies of 30 Kev effective to 400 Kev effective.

III. GEIGER-MUELLER INSTRUMENTS

A. Theory

Essentially, the theory of ion collection in the GM type detector is the same as for the IC instrument except that there is the formation of secondary electrons; that is, primary ions, formed by the incident radiations, are accelerated (given energy) by the high voltage potential and this added energy enables them to produce secondary ion-pairs. The ratio of the total number of secondary ion-pairs produced to the primary ion-pair (Gas Amplification Factor) may be as high as 10^9. For control of the amplification, a quenching gas is introduced. The avalanche, caused by radiation entering the Geiger-Mueller tube, sends a pulse to the indicating unit of the survey instrument. The quenching gas functions to stop the avalanche and makes the GM tube ready for another ionization event. The amplification, inherent in the detector tube, allows a single beta particle or gamma photon to be detected.

B. Physical Description

The principal elements of the portable Geiger counter are: (1) the GM tube with its housing; (2) the electronic circuit; and (3) the meter.

(1) The GM Tube is essentially a glass tube filled with an inert gas at less than atmospheric pressure. The filling gas, usually argon, yields ion pairs (when irradiated). Generally, the tube wall is the cathode and the wire traversing the axis of the tube is the anode. Some tubes have a thin window which admits alpha particles, but these are not used in survey meters to any extent.

Each GM tube has its own characteristic curve of counts/minute versus voltage, which will vary with usage of the tube. The operating voltage must be well up on the Geiger plateau for the proper operation, and it is generally in the range of 1 to 2 thousand volts.

To count efficiently, Geiger tubes must have adequate means for quenching the ionization avalanche started by a particle entering the tube. In survey Geiger counters the tubes are invariably self-quenching; that is, the gas within the tube contains from 10 to 25 per cent vapor of a substance such as ethylene or iso-butane. Quenching gas is decomposed by radiation, therefore a self-quenching tube has a limited lifetime. It is thus obvious, that a Geiger counter should not be left turned on when not in use, especially near a source of radiation.

With a self-quenching tube there is a brief lapse of time from the moment one particle enters the tube until the tube is ready to count the pulse produced by another particle. The initial ionization, the avalanche, the registration of the pulse, and the quenching, all take place in the matter of a few microseconds; then the tube must be cleared in the residual ions. This clearing requires a few hundred microseconds. Dead time becomes important when measuring intense radiation fields.

Most Geiger tube walls are designed so that all but the weakest beta particles may enter. Allowing for the errors due to simultaneous entry, each and every beta particle entering the tube will be counted. Gamma ray counting is not nearly so efficient.

Since one measures each beta particle and each gamma ray that produces ionization within the sensitive tube volume, the instrument is extremely sensitive to radiation, and on the most sensitive scales background levels can be read.

A discriminating shield is provided for the GM tube or probe which when open admits both beta and gamma. With the shield closed only gamma is admitted.

(2) The Electronic Circuit provides the desired voltage to the GM tube, assists in quenching the discharge and receives, amplifies and transforms pulses from the tube to the type of impulse that can be heard in an earphone and registered on a microammeter.

(3) The Indicating Mechanism on most Geiger counters is usually twofold; that is, earphones for aural indication and a meter for visual indication. The meters are in reality microammeters that indicate radiation intensity by a pointer on a scale. The pointer or needle will waver slightly and an average reading should be used. In general, the dial is calibrated either in counts/minute or in milliroentgens/hour, or both. Also, the instrument has a switch for selecting different ranges of sensitivity. For the mr/hr scale, the sensitivities are usually indexed indicating full scale values at a particular switch position, whereas the counts/minute scale is usually marked by x1, x10, x100, or x1,000 of full scale as read on the face of the dial.

Equipment failure is generally due to batteries (some instruments have a battery check in the "on" position), loose connections or faulty GM tube.

C. Operation

The operation of the GM Survey Instrument is essentially the same as that of the IC Survey Instrument. The warmup period is much less critical, and usually 5 to 10 seconds is ample. Care should be taken not to exceed the maximum capacity of the instrument; such excessive exposure will likely damage the GM detecting tube. The GM tube is in operation when in the "on" position and no zero adjustment is possible. It is important to remember that GM survey meters are sensitive instruments and in general do not read high levels of radiation intensity.

D. Calibration

The calibration of GM instruments is the same as for the IC type. The only change that might be noted is that a smaller source of radiation might be used for calibration, since the sensitivity of the GM is much greater than that of the IC instrument.

E. Uses

Survey Geiger-Mueller instruments are useful for low level beta, and gamma ray survey work; with proper modification, they may also be used to monitor for neutrons. Portable GM instruments are available in a variety of types and full scale ranges from .2 mr/hr to 50 mr/hr.

IV. PROPORTIONAL SURVEY INSTRUMENTS

A. Theory

This type instrument derives its name from the fact that it operates in the proportional region of the typical instrument response curve. The probe has an extremely thin window which admits alpha particles to the ionization chamber. The operating voltage is quite high, in the order of 1,500 to 4,000 volts. Gas amplification factors are in the order of 10^5 to 10^6. This instrument can be made to respond only to alpha particles, by choosing the proper operating point in the proportional region, and by circuit adjustment. Alpha particles, since they have the highest specific ionization, give greater pulses than do beta and gamma; thus, by properly adjusting the input sensitivity of the main instrument circuit, we can eliminate all indications from the detecting element except those produced by alpha particles.

B. Physical Description

Proportional survey instruments have three principal components: (1) the ionization chamber; (2) the electronic circuit; (3) the meter.

(1) The Ionization Chamber has walls which serve as one electrode, and wires transversing the chamber which function as the opposite electrode. Such a chamber contains air or gas at normal pressure. A thin window of nylon, etc., allows alpha particles to enter, and the ionization they produce causes secondary ionization proportional to the production of primary ion pairs.

(2) The Electronic Circuit is more complex than that used in other type instruments. It is necessary to control the chamber voltage within fairly narrow limits.

(3) The Meter is marked in counts/minute with several sensitivity scales. The dial is used in a manner already discussed.

C. Operation

The operation of the proportional radiation survey instrument is similar to other instruments. A warmup period of several minutes is usually required to allow the circuit to become properly energized.

D. Calibration

Calibration of proportional meters for alpha contamination is accomplished by means of known quantities of the alpha emitter in question deposited on planchets.

E. Uses

Proportional survey instruments find their greatest application in alpha survey work. In making any alpha survey, the instrument probe must be extremely close to the surface being monitored.

V. SCINTILLATION SURVEY INSTRUMENTS

A. Theory

Scintillation counters depend upon the light produced when ionizing radiation interacts with a phosphor or crystal of certain substances capable of producing this light. The scintillations produced in the phosphor or crystal are then permitted to fall on a photomultiplier tube which converts the light pulses to electrical impulses. These electrical impulses may then be amplified and caused to register on a microammeter.

B. Physical Description

Scintillation type survey instruments have four principal components: (1) the scintillating phosphor or crystal; (2) the photomultiplier tube; (3) the electronic circuit; and (4) the meter.

(1) Scintillating Phosphors may be liquid or crystalline, but for survey work, the crystalline type is, at present, preferable. If one is interested in detecting alpha radiation, a silver activated zinc sulfide screen (similar to the sensitive screen of a television picture tube) is generally used.

For the detection of beta radiation, an anthracene crystal, covered with a thin metal foil to shield out alpha radiation, is preferable; for x- or gamma radiation a sodium iodide crystal is generally employed. When it is desirable to detect neutrons, a secondary reaction must be employed such as the reaction of thermal neutrons with boron in which an alpha particle is released. The alpha particle may then be detected with a ZnS phosphor.

(2) The Photomultiplier Tube picks up light flashes from the phosphor which is in contact with it, and converts these light flashes to electrical impulses. It consists of a photosensitive screen, which emits electrons when light falls on it, and a series of dynodes at a positive potential with respect to the photocathode and with respect to each other. An electron liberated in the photocathode is accelerated to the first dynode, which is about 100 volts positive to the photocathode, where it knocks out additional electrons that are accelerated toward the second dynode. The second dynode is 90 to 100 volts positive to the first dynode and the electrons striking the second dynode produce still more electrons. This multiplication process proceeds through each successive dynode until the electrons reach the anode. From this process, the current amplification is in the neighborhood of 10^9 - 10^{10}.

(3) The Electronic Circuit serves to maintain the voltage across the elements of the photomultiplier tube and to amplify the current impulses from the photomultiplier tube to a magnitude great enough to read on a meter.

C. Operation

Operation of scintillation survey instruments is similar to that of ion chambers and GM instruments. It should be pointed out that the photomultiplier tube of a scintillation instrument will be ruined if exposed to light without first removing the voltage applied to the tube.

D. Calibration

Scintillation instruments may be calibrated in the same manner as is used for a GM or an ion chamber instrument.

E. Uses

As previously pointed out, scintillation devices may be used to detect either alpha, beta, gamma, or x-rays or neutrons simply by placing the proper phosphor in contact with the photomultiplier tube. Scintillation detectors are very sensitive, more sensitive and efficient than GM counters, particularly to gamma radiation. They may be used to detect extremely low levels of activity, as the noise background may be kept much lower than that encountered in the circuiting of a GM or ion chamber instrument. Losses due to dead time in a scintillator are very slight, as light flashes may be produced in many portions of the phosphor at the same time, and the decay time of these flashes is very short; consequently, scintillators are useful for measuring very high radiation intensities.

Personnel Instruments

I. INTRODUCTION

A. It is extremely difficult to measure the absorbed dose or energy deposition in tissues from varieties of radiations. The conditions under which one must work are generally complex, ill defined and irregular. Normally, the best one can hope for is an estimate of the exposure dose to x and γ radiation. To do this requires, usually, not one instrument or method but several. Perhaps the most practical, although less accurate, is the film dosimeter. With adequate definition of radiation conditions and proper control and interpretations, it is possible to evaluate radiation exposures under which personnel must work. This usually gives one enough material from which the absorbed dose might be inferred. Then if such inferences are further substantiated with dosimeter or pocket chamber data and dose rate information from survey instruments one could finally arrive at a rather reliable dose estimate. Therefore the devices of most general importance and which will be discussed below are: (1) film badges, (2) pocket dosimeter, and (3) pocket chambers. Survey meters are covered elsewhere in this manual. Due to the increasing interest in chemical dosimeters, rather than their importance as a personnel monitoring instrument, certain fundamental properties of this device are also mentioned.

II. FILM DOSIMETRY

A. Emulsion

There are a variety of different types of emulsions but the feature in common is a gelatin base with silver halide which is spread on film (cellulose) or glass. The thickness ranges from a few microns to several hundred microns. The most common thickness for nuclear emulsions is from 10 to 25 microns. This corresponds to a mass of about 2 to 5 mg/cm^2. Over the emulsion there is usually a thin protective coating of gelatin sometimes referred to as a T coat (about 0.5 microns). The silver halide grains in most emulsions are silver bromide. The grain size is quite important in determining sensitivity. Of course other constituents of the emulsion as well as developing techniques may modify sensitivity. The constituents or type of the emulsion may be varied depending on the type radiation and the levels to be encountered. For example, if one is concerned with recording the tracks of heavy particles, the grain abundance is adjusted to about 3 times that used in ordinary optical emulsions. Also, certain sensitizers may be added, for example, Boron-10, to greatly increase the sensitivity for thermal neutrons.

B. Theory of Latent Image Formation

The radiation loses energy by setting free or raising the energy of one or more electrons into the conduction energy band of the crystal. Probably more than one ionization event is required. The electrons migrate about the crystal, eventually being trapped in "sensitivity centers" which might consist of impurities or deformities in the crystal lattice. The electrostatic potential set up about the centers results in the accumulation of some silver ions which tend to move freely in the crystal. The number of ions which take part in such migration depends primarily on the temperature. These ions collect around the sensitivity centers due to the attraction for the negative electrons. Ultimately, the ions are neutralized to form the silver atoms. These atoms constitute the latent image of the emulsion and act as a catalyst during the development process to completely convert the grain to silver. Thus, the development process is merely an amplification (of about 10^{12}) of what has already transpired. The probability of a latent image being established increases with an increase in the number of electrons in the sensitivity centers; and this number is proportional to the energy absorbed by the emulsion from the radiation.

C. Limitations

1. ENERGY DEPENDENCE

Figure 1 demonstrates the sensitivity of Ilford Line film to x-rays as a function of their effective energy. The gradual rise in sensitivity as the energy decreases is due primarily to the photoelectric interaction with silver at the lower energies. The decrease in sensitivity below 50 Kev is due to absorption within the outer film covering. Although not shown in Figure 1, the sensitivity would tend to decrease slowly as the energy of the photons is increased above 1 Mev. If the film is used to measure dose from photons of known energy and is calibrated for this energy, the measurements would be reliable. On the other hand, if a wide range of energies are to be encountered, the energy dependence should be reduced such that the response is essentially flat from above 50 Kev to 3 Mev (energies outside of these limits should not be measured in units of roentgens[1]). There are several methods which have been used to reduce the energy dependence. A lower silver bromide content would be of some value but this may affect resolution as well as sensitivity.

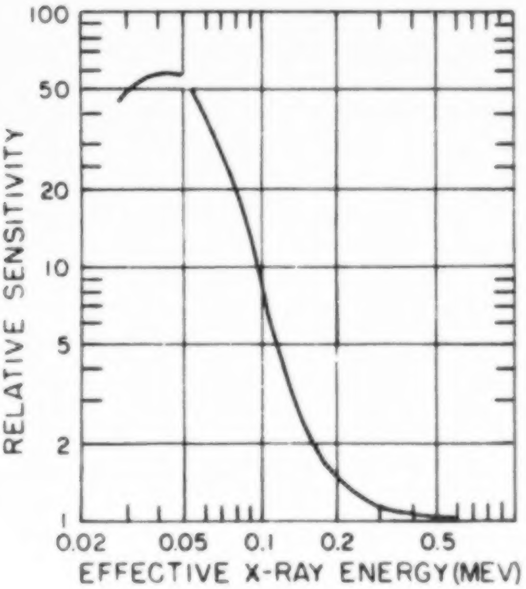


Figure 1

(Reprinted from Hine and Brownell)

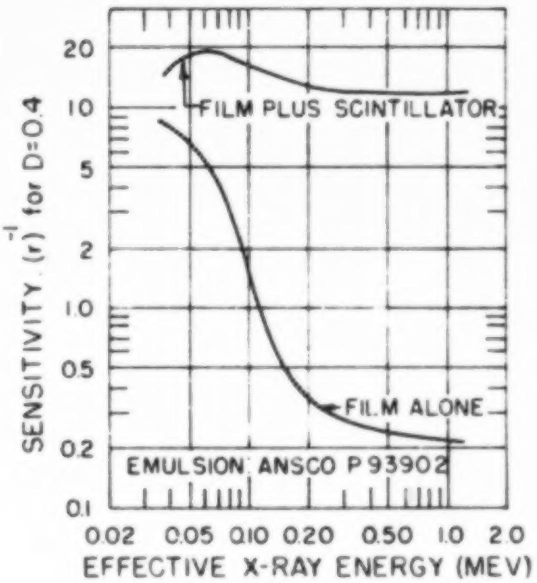


Figure 2

(Reprinted from Hine and Brownell)

Organic scintillators have been used with some success.[2]

The ultra violet fluorescence effectively increases sensitivity as well as improves the energy response. The most common approach has been to utilize a system of filters calibrated to yield information as to the energy of the unknown photons.[3] Such information may then permit an evaluation of the exposure dose.

2. ANGULAR DEPENDENCE

Actually, the dose to material (tissue or emulsion) is independent of the angle of incidence since dose is computed first to an infinitesimal slab then integrated over the entire volume. But film darkening is dependent on angle of incidence. Table 1 illustrates this dependence.

Table 1. Relative Film Badge Sensitivity for γ-Rays Incident at Various Angles

Angle of Incidence	0.11 Mev	0.20 Mev	1.2 Mev
0° (perpendicular incidence)	1.00	1.00	1.00
22.5°	0.87	0.92	0.97
45°	0.46	0.73	0.91
67.5°	0.33	0.45	0.92
90°	0.16	0.41	0.94

3. RATE OF EXPOSURE

The bulk of evidence indicates that the photographic effect is independent of the rate at which that exposure is produced. It is certainly true for intensity ratios of at least 1 to 10,000. There is little knowledge of the effect of microsecond exposures but due to the phenomenon of multi hits (one grain hit by more than 1 particle), it is probably of some significance. On exposures of several days, there is some fading of the film. The fading is influenced by both temperature and humidity. By selection of films of different sensitivity, film dosimeters of wide ranges can be devised. For example, DuPont 502 film can be used from 100 mr to 10 r, while Eastman 548-0 double coat is suitable from 500 to 10,000 r.

4. DEVELOPING TECHNIQUES

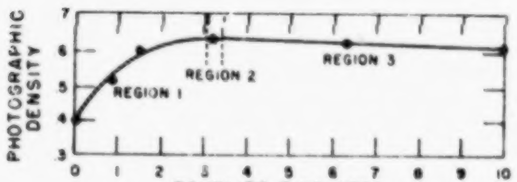
Even slight deviations in the developing time, type, quality and temperature of developer may affect the density in film dosimetry. Therefore, the usual procedure is to develop a set of known standards with each batch of unknown films.

5. MATERIAL SURROUNDING FILM PACKET

Normally, tissue equivalent or air equivalent material is used (lucite, bakelite) having an open window for the detection of beta particles. The thickness of the material must be sufficient to permit electronic equilibrium for the particular energy to be measured. Obviously, in a mixed radiation field, this ideal is difficult and sometimes impossible to achieve. This effect is demonstrated in Figure 3 where the equilibrium thickness is indicated by the dotted line.

D. Neutron Film Dosimetry

1. The measurement of thermal neutron dosage may be accomplished through the



ELECTRONIC EQUILIBRIUM IN BAKELITE, OBTAINED WITH Co^{60} RADIATION

Figure 3

comparison of film densities under cadmium and brass filters. These two filters are designed to attenuate gamma radiation by the same amount. However, because of the (n, γ) reaction induced in the cadmium by the thermal neutrons, the exposure behind the brass when thermal neutrons are present is lower than behind the cadmium. If a wide spectrum of gamma energies are present, a modification of this approach could be used. As an alternative method of measuring neutron dosage, track plates may be used. The fast neutrons interact by (n, p) reactions and the proton recoil tracks are counted and their range is measured. The thermal neutrons interact with the emulsion by N^{14} (n, p) C^{14}; a cadmium filter is usually employed to distinguish between the slow and fast neutrons. The sensitivity for slow neutrons may be increased by adding B^{10} to the emulsion since boron has a very high absorption cross section for thermal neutrons.

III. CHEMICAL DOSIMETRY

Due to the increased use of kilocurie strength sources, there is a growing need for higher range dosimeters. A considerable amount of research is presently underway in this field in an attempt to develop both solid and liquid dosimeters which by color change, photoluminescence, or other alteration, the radiation dose may be estimated.

The principle of such dosimetry is based on the Bragg-Gray Theory which states that *the energy absorbed for unit mass of a substance* (dE/dm) is related to the degree of ionization (J_m) per unit mass produced in a small air cavity within the material.

$$\frac{dE}{dm} = (W)\,(s_m)\,(J_m)$$

W is the energy absorbed per ion pair in the gas; s_m is the relative mass stopping power in the material to that in the gas. If this principle is used to calibrate the material to be used for a dosimeter, a very high degree of accuracy is possible for a variety of different types of dosimeters.

The effect produced in most liquid dosimeters is due primarily to the radiation effect in water. The primary phenomenon in the irradiation of water is the formation of the free radicals H and OH. These may react to form the products, hydrogen, H_2 and hydrogen peroxide, H_2O_2, as well as to re-combine to form water. Hydrogen may then act as a reducing agent on any chemical present capable of being reduced; and H_2O_2 may act as an oxidizing agent on any chemical capable of being oxidized. One of the more useful chemical dosimeters consists of a mixture of water and a chlorinated hydrocarbon such as chloroform, trichloroethylene, or tetrachloroethylene. In such indicators, hydrochloric acid is produced which may be indicated by means of organic dyes which change color depending on the hydrogen ion concentration.

Silver activated phosphate glass is a particularly useful material for a solid chemical dosimeter. The primary phenomenon of radiation interaction is through a process of radiophotoluminescence. As a result of ionizing radiation, new stable photoluminescent centers are created in certain materials. After irradiation, the materials will fluoresce under light of the proper wave length (usually in the ultra-violet), while unirradiated glass will not fluoresce under the same conditions.

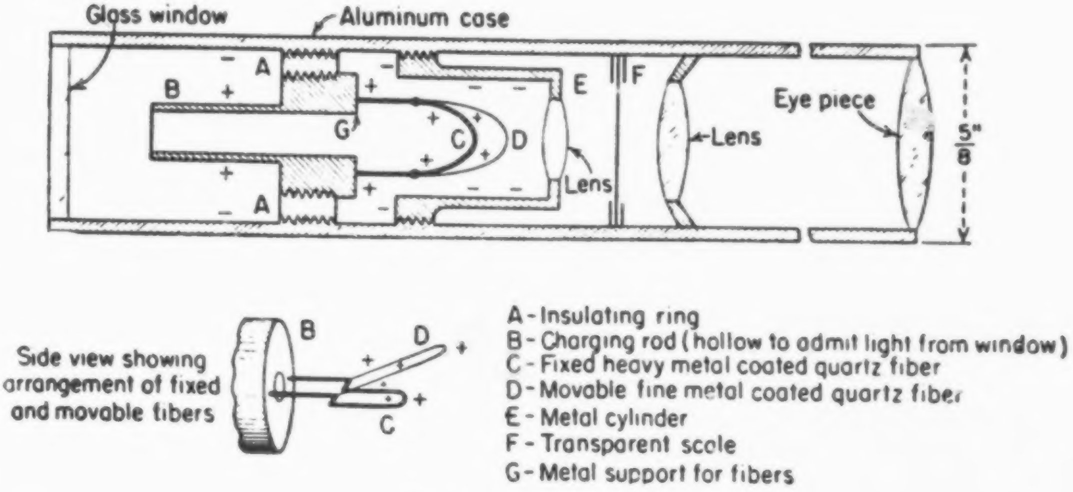
As a final point which should be emphasized, at the present time chemical dosimeters are not very useful in the dose ranges of interest in personnel monitoring. Most chemical dosimeters do not measure exposure doses below about 10 r.

IV. DOSIMETERS AND POCKET CHAMBERS

A. Dosimeters

A pocket dosimeter (Fig. 4) is a chamber containing two electrodes, one of which is quartz fiber loop free to move with respect to its mounting. Like charges are placed on the loop and its moutning, which forces the loop outward from the mount due to the repulsion of like charges. Ionization in the chamber reduces the charge and allows the fiber to move toward its normal position.

An optical system and a transparent scale are all enclosed in the instrument which is about the size and shape of a large fountain pen. The fiber is fused to a metal frame and the microscope is focused on a portion of this fiber. Radiation entering the chamber causes ionization within the sensitive volume. This ionization discharges the electroscope; the distance the fiber moves being proportional to the dose received in the chamber. Instruments of this type can be made sufficiently rugged to withstand the shocks of normal human activity, are small enough to be worn comfortably, and are very useful for measuring integrated exposures. Dosimeters usually are made to have a full scale deflection corresponding to 200 mr, but can also be made with other sensitivities such as 100 mr, 1 r, 10 r, 100 r, etc. The great advantage of this type of instrument is that it can be read at any time without the aid of a supplementary charger-reader by simply holding it up to a source of light and looking into it.


Glass window
Aluminum case
A
B
G
C
D
E
F
Lens
Lens
Eye piece
5"/8
Side view showing arrangement of fixed and movable fibers
A - Insulating ring
B - Charging rod (hollow to admit light from window)
C - Fixed heavy metal coated quartz fiber
D - Movable fine metal coated quartz fiber
E - Metal cylinder
F - Transparent scale
G - Metal support for fibers

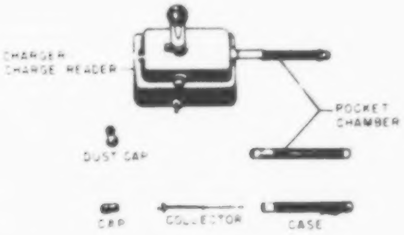
Pocket-dosimeter electroscope.

Figure 4

Reprinted from Ralph E. Lapp and Howard L. Andrews, "Nuclear Radiation Physics", p 199, Prentice-Hall, Inc., 1948.

B. Pocket Chamber

A pocket chamber (Figure 5) has a cylindrical electrode with a coaxial collecting rod well insulated from the rest of the instrument. A charge is placed on this rod. Ions formed in the chamber collect on the rod and reduce the previously placed charge. A pocket chamber differs from the dosimeter mainly in that the electroscope portion (the quartz fiber mechanism and optical system) is in an external unit. This means that the chamber must be charged and read with a separate unit called a "charger-reader". The pocket chamber is similar in size and shape to a fountain pen. The change in charge is measured on a scale that may be calibrated in milli-roentgens. The advantage of this unit is its low cost when compared to a self-reading dosimeter.



POCKET CHAMBER AND CHARGER-CHARGE READER

Figure 5

C. Operation Characteristics

A small amount of dust or lint on an insulator of one of these instruments can be enough to discharge it completely. Therefore, they must be kept clean. Dropping or sudden jarring will also sometimes discharge the instrument.

Insulator "soak-in" is a phenomenon regularly encountered in the operation of these chambers. When an instrument has been out of use for some time and is charged, a rather rapid discharge may be noted. This is due to the penetration of part of the charge into the insulator. It is not leakage in the ordinary sense because it practically disappears if the instrument is kept charged for a day or so. To eliminate this, the instrument should be charged a day or more before it is used. Generally these instruments are designed to respond to x- or gamma radiation, as the walls are too thick to admit beta or alpha radiation.

The chambers are calibrated by the manufacturer to read exposure in milli-roentgens, or roentgens, but they show some energy dependence in their sensitivity and may, therefore, read erroneously. They are relatively energy independent at high energies (greater than 200 Kev) but it is best to know what energy radiation is involved for correct interpretation. Calibration curves can and often should be made for pencil chambers.

V. SUMMARY

Film badges, dosimeters, and pocket chambers are the principal personnel monitoring instruments. Each has specific advantages. Film badges provide a permanent record of exposure, but require more work to obtain readings than the ionization instruments. Dosimeters are especially useful to determine exposure at any time while continuing to act as a monitor without further attention. Pocket chambers are quite simple and therefore represent an inexpensive means of measuring exposure.

There are other personnel monitoring instruments which have limited usefulness in this field. Hand and foot counters are only applicable in permanent locations such as laboratories. Pocket alarms are impractical due to their size and weight, when carried on the person. At the present time, chemical dosimeters are too insensitive for measuring low-level, chronic type of personnel exposures; however they offer a valuable adjunct in estimating dose intensities during radiation accidents and of course there are a variety of research and industrial applications.

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GPO 876-257

Laboratory Instruments for Alpha and Beta Counting

I. INTRODUCTION

Several basic laboratory instruments have been designed for counting low-level amounts of alpha and beta emitters. These devices are extremely sensitive, and are capable of determining levels of activity in the micro- to picocurie range.

I INSTRUMENT COMPONENTS

All laboratory instruments have two basic components--a detection medium and associated electronic circuitry.

A. Detection Media

Three types of detection media--gases, solids, and liquids--are currently used for alpha and beta counting. Detection in any of these media is based upon the radiation's ability to cause ionization and excitation.

1. GASES

Typical gases used as detection media are argon, xenon, and methane. If a chamber is filled with one or a mixture of these gases, and two high-voltage electrodes are immersed in the gas, the ions produced in the gas will be collected at electrodes of the opposite electrical sign. The flow of ions is detected in the circuitry of the laboratory instrument.

2. LIQUID AND SOLID MEDIA

Ionization and excitation processes in certain crystals, plastics, or liquids are followed by de-ionization and de-excitation processes resulting in the emission of small flashes of light. A photomultiplier tube converts these scintillations into electrical pulses, which are fed into the circuit.

B. Electronic Circuitry

Electronic current pulses from the detection medium are fed into an electronic circuit, where they may be observed as total ionizing events.

The major components of such circuitry include:

1. PREAMPLIFIER

This unit amplifies the detector pulse for later electronic stages. It also serves as an impedance matching device between the detector media and the electronic circuitry. Transistors have greatly simplified and improved the reliability of the new preamplifier circuits.

2. DISCRIMINATOR

The discriminator circuit may be set so that the scaler responds only to pulses larger than a predetermined size, or within a given energy range. Thus, a discriminator permits selective counting of mixed radiations.

3. SCALER

The scaler component functions as an electronic adding machine, and it is commonly used with an attached mechanical register and timer. The Eccles-Jordan trigger circuit, with modifications, forms the basis for most scaling circuits used in counting work. These modified circuits are referred to as multivibrators. Scalers may be designed with either binary or decade systems.

The binary system employs a "scale of two" and has a number of selective scaling factors--usually 8, 16, 32, 64, 128, 256, and 512--with neon interpolation lights. The total count is obtained by multiplying the register reading by the appropriate scaling factor and adding the sum count indicated by the interpolation lights. The decade system uses four "scales of two" electronically connected in such a fashion that for every ten input pulses, one output pulse is registered. Counts registered by the decade system may be read directly.

4. REGISTER

The register used with scaling circuits consists of either a relay-operated mechanical register, or a recording device which indicates the number of pulses received from the last scaling stage.

5. TIMER

The most convenient method of counting is with an attached timer, which shuts the scaler off after a predetermined time of operation. Other timers, a manual type and a preset count type, are also occasionally used.

III. END-WINDOW GEIGER-MUELLER COUNTER

This instrument, primarily designed for counting beta radiation, has been widely used in radiological laboratories. Detection efficiencies range up to 25-30 percent. The instrument can also detect alpha radiations if their energy is not totally absorbed by the air space between the sample and the detector or by the detector window itself.

A. Detector Construction

The end-window Geiger-Mueller detector is usually a cylindrical glass tube (about 2½ inches long and 1½ inches in diameter) with a plastic base to which four pins are attached. (See Figure 1.) A plastic rim on the other end of the tube holds in place a thin mica window of 1 to 3 milligrams/square centimeter thickness. The tube is filled with helium or argon and a quenching gas, such as alcohol vapor. The central electrode (anode) consists of a thin wire about 0.01 of an inch in diameter. The outer electrode (cathode) consists of a layer of copper or silver plating on the glass wall of the tube.

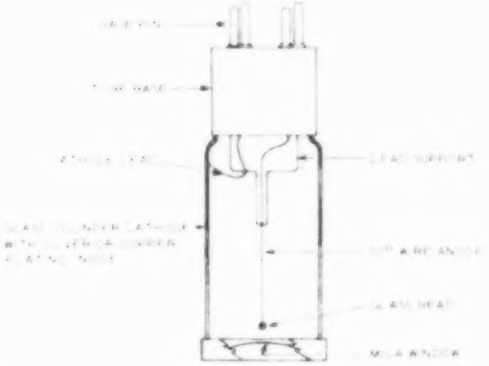

Figure 1.--End-Window Geiger-Mueller Counter Tube

B. Detector Operation and Shielding

Each end-window G-M tube has its own characteristic curve. This curve is established by placing a source opposite the tube window; if the counter voltage is then increased by 25 to 50 volt steps, with counting rates being established for each voltage setting, a characteristic curve may be plotted. The usual curve has a plateau section (see Figure 2) wherein the counting rate shows only a small change with voltage variations. The tube operating voltage is usually selected about one-third the plateau length from the threshold to the breakdown voltage.

The G-M tube is normally placed in a brass-lined lead shield or "pig" to cut down the background count. A typical lead shield weighs several hundred pounds.

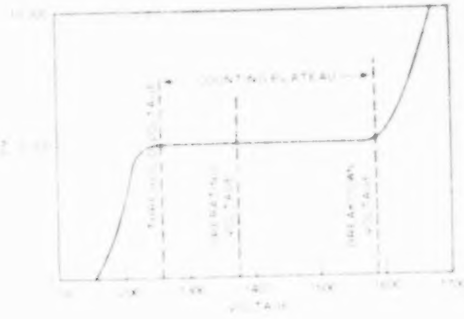

Figure 2.--Characteristic Curve of Typical G-M Tube

C. Counter Circuitry

The electrical pulse from the G-M tube is fed through a series of modified multivibrator units in the scaler and is recorded with a register and neon scaling lights.

IV. INTERNAL PROPORTIONAL COUNTER

This instrument has been designed to count alpha and/or beta particles. Typical efficiencies for counting alpha radiations may be as high as 50 percent, while efficiencies for beta radiations may vary between 55 and 75 percent.

A. Detector Construction

The internal proportional counter detector is a chamber usually designed as a steel hemisphere. (See Figure 3.) The bottom of the chamber supports the sample dish and is called the piston. It is joined to the hemisphere (when counting) at a rubber O-ring junction. A plastic insulator holding a tungsten wire loop anode of 0.001-inch diameter is inserted in the top of the hemisphere. The hemisphere wall functions as the cathode, and contains a counting gas inlet and outlet. Since atmospheric air will "poison" an alpha or beta count, the chamber is purged with a counting gas mixture (usually 90 percent argon, 10 percent methane) before a count is made. This same gas slowly flows through the chamber during a counting period; the positive pressure inside the chamber should insure against air leaks at the O-ring seal.

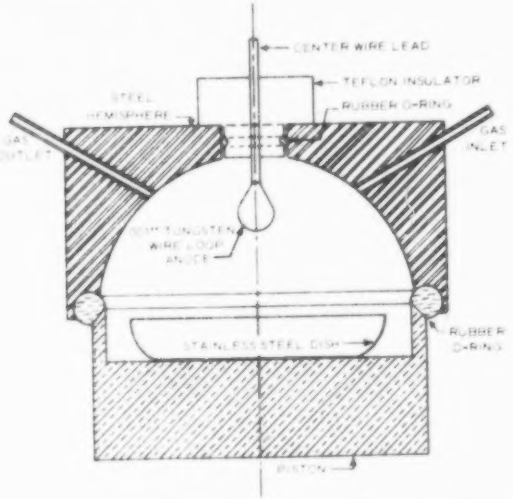

Figure 3.--Internal Proportional Counter Chamber

B. Detector Operation

The internal proportional detector has two characteristic curves (see Figure 4), each with a plateau similar to that for the G-M tube. The first plateau, called the "alpha plateau," has a typical voltage range of approximately 900-1300 volts. The second plateau, called the "beta plateau," has a typical voltage range of approximately 1600-2000 volts. The midpoint of each plateau is normally selected as the operating voltage. When the instrument is adjusted at the alpha operating voltage, only alpha particles are counted. When the instrument is adjusted at the beta operating voltage, both alpha and beta particles are counted. To count only beta particles, the chamber may be operated on the beta plateau and an aluminum foil of 8 mg/cm^2 thickness placed on top of the sample to shield out the alpha particles. Of course, an indeterminate fraction of the lower energy beta particles will

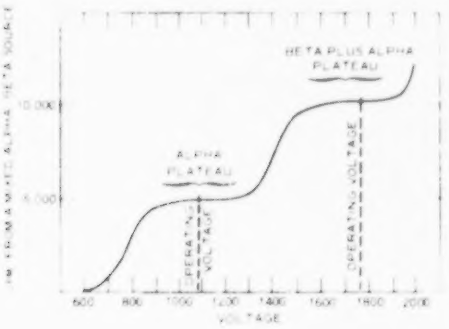

Figure 4.--Operating Curve for a Proportional Counter

also be stopped in this shield. An equally precise and somewhat more convenient procedure is simply to count the alphas on the alpha plateau, the alphas-plus-betas on the beta plateau, and subtract to obtain the net beta count. (This method would be exact, except for the alpha plateau slope of a few percent per hundred volts, which causes each alpha to produce about 1.1 to 1.2 counts on the beta plateau. In practice, this slope is easily measured and an exact correction to the data may be made.)

C. Counter Circuitry

A preamplifier and a discriminator characterize the typical prescaling circuitry in an internal proportional counter. At the alpha operating voltage, the discriminator permits selective response to alpha particles with little or no response to high intensities of beta and gamma radiation. An alpha particle may produce a total of 10^8 primary and secondary ionizations in the chamber, while a single beta particle or gamma ray may produce a total of 10^5 ion pairs. If the discriminator is adjusted to respond only to a minimum of 10^8 ionizations for each initial ionizing event, a count of only alpha particles will be recorded.

Internal proportional scaler units are essentially the same as those used for end-window G-M counting. Typical backgrounds for unshielded 2-inch chambers are 0.1 c/m on the alpha plateau and 50 c/m on the beta plateau.

V. SOLID SCINTILLATION COUNTERS FOR ALPHA AND BETA COUNTING

The principles and construction of scintillation counters are discussed in TP-78, "The Gamma Scintillation Spectrometer." The only difference between alpha and beta scintillation counters is the composition of the detection medium.

A. Alpha Scintillation

A thin layer of silver-activated zinc sulfide is usually used to detect alpha particles. This material emits light flashes when bombarded with alpha radiation.

B. Beta Scintillation

Anthracene is commonly used as a beta particle scintillation medium.

VI. LIQUID SCINTILLATION COUNTERS FOR BETA COUNTING

These systems provide effective means of counting beta emitters of both low and high energies. The radioactive material is dissolved in a suitable solvent, to which selected scintillators have been added. The energy deposited in the mixture is converted to light flashes; the intensity is proportional to the energy of the incident radiation. This signal is electronically amplified and recorded on a scaling device.

For more detailed information on the subject, see TP-416, "Liquid Scintillation Counters."

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Counting Efficiency

I. SAMPLE PREPARATION

The type of radiation to be detected and the chemical and physical form of the sample determine the type of detector to be used. For common types of detectors used, ideally the sample should be weightless (to negate self-absorption losses) and should be either a point source or spread uniformly over a given area (to facilitate replication of geometry).

With alpha or soft beta radiation particularly, thin deposits of sample are preferred. In some instances, it may be possible to concentrate the radioactive material by chemical or physical means such as electro-deposition, precipitation, evaporation, etc. In other instances, notably with carbon-14, the material may be converted to a gas which can be counted in an internal-type detector.

With gamma emitters, absorption effects are much smaller, which allows thicker samples to be counted. Using high efficiency gamma detectors, soil or liquid samples can be counted directly.

II. ABSOLUTE COUNTING

Absolute counting requir [illegible] valuation of the observed counting r [illegible] ns of the disintegration rate. The [illegible] served counting rate to distintegration [illegible] known as the counter efficiency which can be determined for any instrument by either of two methods: (1) by measurement of the individual factors (geometry, absorption, and scatter) which determine the efficiency, or (2) by comparison with a known standard source of radioactivity.

A. Measurement of Indivudual Factors

By this method, the efficiency is equal to (geometry factor) x (scatter factor) x (absorption factor). These factors are defined and explained as follows:

1. GEOMETRY

The prime factor affecting the efficiency of any detector is the position of the sample relative to the sensitive volume of the detector. The geometry is defined as that percent of solid angle about a source which is subtended by the sensitive volume of the detector. In the case of an end-window G-M counter, increasing the vertical separation between the sample and the face of the tube would lower the geometrical efficiency by decreasing that solid angle. The geometry of end-window counters usually ranges from 1 to 30%. For an internal counter, where the sample comprises a portion of the floor or one wall of the counting chamber, the geometry is fixed at 2π, i.e. 50%.

The geometry (G) of an end-window detector using a point source may be calculated as follows:

$$G = 0.5\,(1 - \cos\alpha)$$

where: $\alpha = \tan^{-1} r/d$, r is the radius of the counter window and d is the distance between source and window.

The geometry may also be determined experimentally using a known source mounted such that scatter and absorption effects are negligible.

2. SCATTERING

The most important of the scattering effects is "backscattering" i.e., the reflection by the source support material of radiations emitted by the sample away from the sensitive detector volume. Backscatter is defined as the ratio of the number of counts achieved from a sample mounted on any given material to the same sample suspended freely in air.

This scattering effect is a function of three variables: the energy of the radiation, the thickness of the source support, and the material employed in the source support.

The higher the energy of the radiation, the more scattering there will be. The higher the atomic number of the sample support, the higher the backscatter factor will be. Up to a certain thickness (known as the saturation thickness), the thicker the sample mount, the greater the backscatter will be. Backscatter factors may range as high as 1.6 for materials such as silver and platinum.

There are other scattering actions which take place with any counting arrangement including scattering from the source support structure, the walls of the lead shield and the walls of the tube itself, but these effects are minor when compared with backscatter, and will be constant for any sample in the same position.

To determine the backscatter factor, counting rates are compared using essentially weightless backing (sample mounted on a thin plastic film) and using the sample mount and support of interest. Experimentally determined backscatter factors have been reported in the literature and these are applicable for similar counting arrangements.

3. ABSORPTION

There are two absorption effects which are important. The first is absorption taking place in the source itself. The second is that taking place in the air and detector window.

Self-absorption is most significant in the cases of alpha emitters, since extremely small amounts of material thickness are sufficient to completely absorb the alpha radiation. Carbon-14 with its low beta energy shows significant self-absorption effects at thicknesses of one-milligram per square centimeter, while P^{32} (which has a maximum beta energy some ten times as large) will not be significantly affected until the sample thickness reaches values of from 15 to 20 milligrams per square centimeter.

Absorption taking place in the air and detector window would apply primarily to an end-window G-M counter in which case the thickness of air and window would have to be determined. The "equivalent window thickness" can be calculated by adding to the thickness of the tube window (given by the manufacturer) the thickness of air separating the source and the tube. To determine the latter in mg/cm^2, multiply the distance in cm by 1.2.

To determine the absorption factor (ratio of count rate without absorption to that with absorption), samples are prepared using a fixed amount of absorbing material. A plot of the count rate versus sample thickness can be extrapolated to zero thickness.

B. Comparison with Standard Source

A simpler method for determining counting efficiency is merely to compare the observed counting rate with the known disintegration rate of a standard reference source. Such reference sources are commercially available for a wide variety of radioisotopes. In this way the efficiency can be measured for similar samples on similar mountings placed in a similar shelf position. This method is usually employed in that it is easier, quicker, and perhaps more reliable.

III. COMPARATIVE COUNTING

Comparative counting is usually encountered in tracer studies where the objective is to determine relative source intensities rather than absolute intensities. Normally, sample preparation and mounting are identical for various samples counted so that there is no need to correct counting rates to absolute disintegration rates.

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Nuclear Counting Statistics

I. INTRODUCTION

The statistical nature of radioactive decay stems from the fact that radioactive atoms decay at completely random intervals of time. This concept will be considered here; some basic principles of statistics and techniques to handle data acquired by nuclear counting instruments will be mentioned briefly.

II. THE STATISTICAL NATURE OF RADIOACTIVE DECAY

A. Random Time Intervals

Why is radioactive decay statistical? If the decay (disintegration, α, β, or γ emission . . .) rate of a particular radioisotope were determined to be one decay per minute, it might appear straightforward that in one minute, one count would be detected, two minutes, two counts, and so forth. However, breaking the *time* interval down further (into seconds) would the group of radioactive atoms decay once in the first ten seconds, after thirty seconds, or once every sixty seconds? Atoms decay randomly and could decay over any time interval. The probability for an atom to decay is independent of time. Even with 100% efficiency detection equipment, there will always be uncertainty in a nuclear measurement because of this random decay process.

Note that the statistical nature of radioactive decay has nothing to do with the statistical fluctuations associated with the electronic detecting equipment. It is solely a phenomenon of radiation. It is an error of definition.

B. Radioactive Decay Law

In 1905 the law of radioactive decay was deduced by Schweidler and was shown to be exponential in nature. This law, expressed mathematically is:

$$N = N_0 e^{-\lambda t},$$

Where:

N = nuclei remaining after a time interval, t
N_0 = nuclei at some original time
e = base of natural logarithms, 2.7183 . . .
λ = decay constant for the particular radioactive element, and
t = elapsed time.

This law of radioactive decay was originally derived with only *three* assumptions: (1) The disintegration of an atom is subject to the laws of chance. (2) The probability of an atom decaying in a certain time interval is independent of the past history of the atom. And (3) this probability is the same for all atoms of the same element.

A rough analogy could be drawn from tossing a coin. There are two possible outcomes: heads or tails (either the atom decays or it does not decay). Whether waiting 2 minutes or 3 centuries to toss the coin, the probability of receiving a head is still the same (50%). If an atom has not decayed for 2 minutes or 3 centuries, the probability that it will decay in the next second is still the same as previous time intervals (small but constant). And finally, no matter which of the billions of minted coins is flipped, the probability of receiving a head is the same for all (the probability of an atom decaying is the same for all similar atoms).

III. STATISTICAL PRINCIPLES

The fundamental frequency distribution governing random events is the binomial distribution. Binomial literally means two numbers and, in radioactive decay, pertains to the two possible outcomes. Either the atom decays or it does not decay. When the probability of an event occurring is small and constant (as in the case of disintegrating atomic nuclei), the Poisson distribution (a limiting case of the binomial) can be used to describe the frequency of occurrence. Both the binomial and Poisson distributions pertain to discontinuous variables (such as particle counting rates) which take on successive whole number integral values. For a moderately large number of events (> 30), the "normal" or Gaussian distribution provides an adequate approximation easily applied to nuclear counting statistics. The statistical theory of errors is ordinarily based on this normal distribution.

A. Normal Distribution

The familiar "bell-shaped" curve is a plot of the normal distribution.

Figure 1 shows the frequency of occurrence of some event, x, plotted against the numerical value measured for that event. If student grades were graphed, the y-axis would be the number of students (frequency of occurrence) receiving grades while the x-axis would be the numerical or letter grade. The peak and center section would be the *average* or C grades and contain the majority of students. The wings, or rare occurrences, coincide with the A and F students.

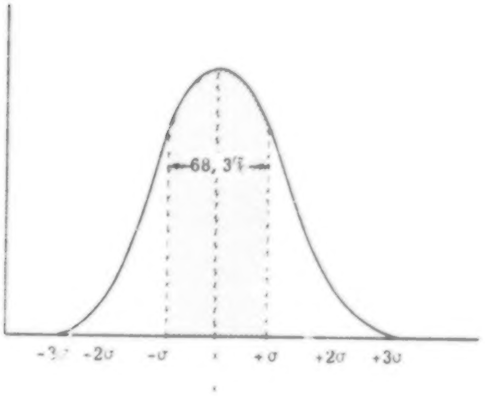

Figure 1. Frequency of Occurrence

When considering nuclear counting statistics, the x-axis would depict the different counts or count rates measured while the y-axis would pertain to the number of times these different counts or count rates were observed (frequency of occurrence or observation).

B. Useful Terms

Three terms commonly used when describing a normal distribution are the "mean," the "standard deviation," and the "confidence level."

The *Mean*, X, is the average value, the center (peak) of the distribution. If two counts of 5 and 7 were obtained, the mean would be 6, [(5 + 7) ÷ 2].

The *Standard Deviation*, σ, is simply a parameter describing the uncertainty of a measurement. It concerns the distribution of *deviations* (X ± σ) from the mean (X). If the total area under the curve of Figure 1 is considered 100%, then the individual observations that deviate from the mean or average value (X), by more than *one* standard deviation (± 1σ), should be about 32% (if the sampled population follows a normal distribution). Thus, 68% of the individual observations should lie within the band X ± σ.

What does this mean? If a radioactive sample were counted many times, 68% of the observations (counts) should fall close to the mean value (within 1 standard deviation), while 32% of the observations should be outside the 1 standard deviation.

Note that the standard deviation estimates the uncertainty (error) in a measurement and is not a discrepancy (false measurement). The uncertainty in counting radioactive elements comes from the statistical or random nature of radioactive decay.

If 64 counts were measured, the standard deviation would be: $\sigma = \sqrt{64} = 8$. Thus 64 ± 8 indicates something of the certainty of the measurement.

The *Confidence Level, or C.L.*, indicates the certainty of or confidence the experimenter has in a measurement. Since 68% of the area under the normal curve falls within ± 1σ standard deviation of the mean value, X, the use of ± 1σ shows a level of confidence of 68%. A standard deviation of ± 1.64σ results in a 90% confidence level; ± 1.96σ is commonly used and gives a confidence level of 95%. For this example:

$$64 \pm 1\sigma = 64 \pm 8 \text{ for a 68\% C.L.}$$

$$64 \pm 2\sigma = 64 \pm 16 \text{ for a 95\% C.L.}$$

Thus, 68% of the time (if the counting experiment were reproduced many times) a value between 56 and

72 should be observed, while 95% of the time, the count should fall between 48 and 80 (assuming, of course, that the sampled population follows a normal distribution).

IV. STATISTICAL TECHNIQUES

A. Count Rate

In nuclear counting experiments it is of more interest to know the count *rate* than the number of counts. The count rate (R) is simply the number of counts (n) divided by the counting time (t), or

$$R = n/t\ .$$

The uncertainty (error) in the count rate is equal to $\sigma = \sqrt{n} \div t$. This calculation assumes a negligible counting-time error. This is justified, since modern clocks and timers are so accurate that the uncertainty in the time is insignificant.

Since $R = n \div t$, $n = Rt$ can be substituted into the rate error equation

$$\sigma = \frac{\sqrt{n}}{t} = \frac{\sqrt{Rt}}{t} = \sqrt{\frac{R}{t}}$$

Example

If 5000 counts were observed during a 4-minute counting period, what is the count rate at 95% confidence level?

$$n_S = 5000 \text{ counts}$$

$$t_S = 4 \text{ minutes}$$

$$R_S = \frac{n}{t} = \frac{5000}{4} = 1250 \text{ cpm}$$

$$\sigma_S = \sqrt{\frac{R_S}{t_S}} = \sqrt{\frac{1250}{4}} = \sqrt{312} = \pm 17.5\ .$$

Using $2\,\sigma_S$ for ~ 95% C.L., $2\,\sigma_S = 2\,(\pm\,17.5) = \pm\,35$. Therefore, count rate, $R_S = 1250 \pm 35$ cpm at 95% C.L.

B. Percent or Relative Error

Frequently the count rate is represented ± its relative or percent error. This is expressed

$$R_S \pm \frac{\sigma_S}{R_S} \times 100\% \text{ (for 68\% C.L.)}$$

and

$$R_S \pm \frac{2\sigma_S}{R_S} \times 100\% \text{ (for 95\% C.L.)}.$$

In the example:

$$\frac{2\sigma_S}{R_S} \times 100\% = \frac{2(17.5)}{1250} \times 100\% = 2.8\%\ .$$

Therefore:

$$R_S = 1250 \pm 2.8\% \text{ (at 95\% C.L.)}\ .$$

C. Background Considerations

In general, there is always a background counting rate, R_B, which must be subtracted to yield the actual sample counting rate, R_S. For instance, if 128 cpm were the count rate (R_{S+B}), of a certain sample, and the background rate (R_B), was 72 cpm, the net sample count rate would be:

$$R_S = R_{S+B} - R_B = 128 - 72 = 56 \text{ cpm}$$

What would be the uncertainty (error) in the net sample count rate? Is it $\sqrt{56 \div t}$? No! Since the theory of propagation of errors must be used, the error (σ_S) will be the *square root of the sum of the squares of the individual errors.*

$$\sigma_S = \sqrt{(\sigma_{S+B})^2 + (\sigma_B)^2}$$

Example:

What is the error at a 95% C.L. for the problem if the counting times for both sample and background were 2 minutes?

$$R_{S+B} = 128 \text{ cpm}$$

$$t_{S+B} = 2 \text{ minutes}$$

$$\sigma_{S+B} = \sqrt{\frac{R_{S+B}}{t_{S+B}}} = \sqrt{\frac{128}{2}} = \sqrt{64} = 8$$

$$R_B = 72 \text{ cpm}$$

$$t_B = 2 \text{ minutes}$$

$$\sigma_S = \sqrt{\frac{R_B}{t_B}} = \sqrt{\frac{72}{2}} = \sqrt{36} = 6$$

$$\sigma_S = \sqrt{(\sigma_{S+B})^2 + (\sigma_B)^2} = \sqrt{(8)^2 + (6)^2}$$

$$= \sqrt{64 + 36} = \sqrt{100} = 10$$

$$R_S \pm 2\sigma_S = 56 \pm 2\ (10) = 56 \pm 20 \text{ at } 95\% \text{ C.L.}$$

A handy formula for net sample count-rate error is:

$$\sigma_S = \sqrt{\frac{R_{S+B}}{t_{S+B}} + \frac{R_B}{t_B}} .$$

Substituting these values:

$$\sigma_S = \sqrt{\frac{128}{2} + \frac{72}{2}} = \sqrt{64 + 36} = \sqrt{100} = 10 .$$

D. Background and Counting Times

It is often not necessary to have a background counting time equal to the sample counting time. Another useful formula to determine the most efficient use of counting times is:

$$\frac{t_{S+B}}{t_B} = \sqrt{\frac{R_{S+B}}{R_B}} .$$

Example:

If the sample counting rate were 400 cpm and the background 25 cpm, how much longer should the sample be counted than the background?

$$R_{S+B} = 400 \text{ cpm}$$

$$R_B = 25 \text{ cpm}$$

$$\frac{t_{S+B}}{t_B} = \sqrt{\frac{R_{S+B}}{R_B}} = \sqrt{\frac{400}{25}} = \sqrt{16} = 4$$

$$t_{S+B} = 4\, t_B$$

The sample should be counted 4 times as long as the background counting time. Thus, if 1 hour were available for the count, 12 minutes would be spent counting background and 48 minutes counting the sample.

This formula is utilized only for "one-shot" counting operations; for each sample, a separate sample count rate and background count rate should be made. In usual counting-room practice, however, it is not necessary to take a background count rate for every sample; background counts need be taken only 2 or 3 times a day. The formula is not applicable without modification.

Example:

What is the error at a 95% C.L. for the problem if the counting times for both sample and background were 2 minutes?

$R_{S+B} = 128$ cpm

$t_{S+B} = 2$ minutes

$$\sigma_{S+B} = \sqrt{\frac{R_{S+B}}{t_{S+B}}} = \sqrt{\frac{128}{2}} = \sqrt{64} = 8$$

$R_B = 72$ cpm

$t_B = 2$ minutes

$$\sigma_S = \sqrt{\frac{R_B}{t_B}} = \sqrt{\frac{72}{2}} = \sqrt{36} = 6$$

$$\sigma_S = \sqrt{(\sigma_{S+B})^2 + (\sigma_B)^2} = \sqrt{(8)^2 + (6)^2}$$

$$= \sqrt{64 + 36} = \sqrt{100} = 10$$

$$R_S \pm 2\sigma_S = 56 \pm 2\ (10) = 56 \pm 20 \text{ at } 95\% \text{ C.L.}$$

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Substituting these values:

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V. PRACTICAL AIDS TO COMPUTATION

For a direct numerical approach, using the formulae discussed, a slide rule is usually accurate enough for squaring and taking square roots. Computation may be considerably simplified by the use of Jarrett's nomographs, the most useful of which are reproduced on pages 115-118 of the Radiological Health Handbook (Rev. 1970). These cover counting rates from 1 to 100,000 cpm and counting times from 1 to 1,000 minutes, a range more than adequate to encompass the usual counting procedures. With a nomograph, a straightedge is used to connect the observed count rate (cpm) with the duration of the count (minutes), and the error at the 90% or 95% C.L. is read off the intersection with the center line, on the left or right side respectively. In the example previously solved, the nomograph method gives a 95% C.L. error of ± 34.7 cpm for a sample which produced a count rate of 1250 cpm for 4 minutes. For background subtraction, the error in the background determination would be found in the same way, using the nomograph, and the individual errors from sample and background would be combined by squaring, adding the squares together, and extracting the square root.

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Correlation of Units of Activity and Exposure

I. INTRODUCTION

Disintegrations of atomic nuclei yield a variety of radiations. These may range from a single alpha particle, a single beta particle, or a single gamma photon to various combinations of either an alpha or a beta particle and one or more gamma photons. Each of these several radiation types may, in turn, possess a wide range of characteristic energies. Thus because the curie, by definition, refers only to the rate of disintegration, there is no simple rule for relating activities in curies and resulting dose rates in roentgens per unit time.

II. THEORETICAL RELATIONSHIP

If certain facts are known, the dose rate resulting from exposure to external sources of radioactive material can be calculated.

A. Assumptions

To simplify the calculations described herein, several assumptions will be made:

1. The source of radioactive material is made up of only one radionuclide.

2. The radioactive material is confined to a small volume and thus serves essentially as a point source.

3. The source is encapsulated and therefore only its gamma radiations need be considered in exposure calculations.

B. Basic Data

Having made these assumptions, the basic data required for relating curies and roentgens are:

1. Identity of the Source - Each specific radionuclide has its own characteristic decay scheme; this includes the number as well as energy of its emitted radiations. Decay data for several selected radionuclides are shown in Table I.(1)(2)

2. Source Strength - Knowledge of the disintegration rate of the source is necessary so that the rate of emission of each of its several characteristic radiations can be calculated.

3. Distance from Source to Point of Exposure - For a source confined to a small volume, the exposure varies inversely as the square of the distance between the source and the point of exposure.

4. Absorption of Emitted Radiations - Radiation absorption is energy dependent. Once the types and energies of the emitted radiations are known, information concerning their absorption can be secured.

C. Example

To exemplify the theoretical approach, consider the following specific example:

1. THE SOURCE CONTAINS Co^{60}

As is shown in Table I or as may be determined from the "Table of Isotopes" in the Radiological Health Handbook, (3), Co^{60}

Table I

Decay Data For Several Selected Radionuclides(1)(2)

Nuclide	Type of Decay	Particles		Gamma Transitions	
		Energy (Mev)	Frequency (% of disintegrations)	Energy (Mev)	Frequency (% of disintegrations)
Po^{210}	Alpha	5.305	100	-----	-----
P^{32}	Beta	1.710	100	None	None
Cs^{137}	Beta	0.514 1.17	92 8	0.662*	92
Co^{60}	Beta	0.314 1.48	≅ 100 0.01	1.173 1.332	100 100
Fe^{59}	Beta	0.126 0.271 0.462 1.561	1 46 53 ≅ 0.3	0.145 0.191 0.336 1.099 1.29	0.8 2.5 0.3 56 44

* Actually emitted by Ba^{137m}, the short lived daughter of Cs^{137}.

emits two gamma photons per disintegration. These have energies of 1.17 and 1.33 Mev respectively.

For the purpose of simplicity, assume that the two gamma photons emitted per disintegration have an average energy of:

$$\frac{1.17 + 1.33}{2} = 1.25 \text{ Mev.}$$

2. THE SOURCE STRENGTH IS ONE CURIE

One curie by definition is that quantity of any radionuclide having a decay rate of 3.7×10^{10} disintegrations per second.

Thus the source under question may be assumed to be releasing 2 photons per disintegration, 3.7×10^{10} disintegrations per second, or 7.4×10^{10} photons per second, each with an average energy of 1.25 Mev.

3. THE SOURCE TO EXPOSURE DISTANCE IS ONE YARD

If it be assumed that the radiations from the source are emitted uniformly in all directions, the number passing through each square centimeter of surface area of an imaginary sphere one yard in radius with the source at its center can be calculated as follows:

a. Surface area of sphere

$$\text{Area} = 4\,\pi\,R^2 \qquad \text{where } R = 1 \text{ yard}$$

$$\text{Area} = 4\,\pi\,[(1 \text{ yard})(36 \text{ inches/yard})(2.54 \text{ cm/inch})]^2$$

$$\cong 10^5 \text{ cm}^2$$

b. The number of photons passing through each square centimeter of surface area per second equals

$$\frac{7.4 \times 10^{10} \text{ photons/second}}{10^{5} \text{ cm}^{2}}$$

$$= 7.4 \times 10^{5} \text{ photons/cm}^{2}\text{-sec.}$$

4. ABSORPTION OF EMITTED PHOTONS

The attenuation of gamma radiation passing through matter may be described by the equation:

$$I = I_o e^{-\mu x}$$

where:

I_o = initial flux of photons

μ = linear absorption coefficient

x = thickness through which the photons have passed

I = final photon flux

The number of photons absorbed in passing through a given thickness of matter is:

$$I_o - I = I_o - I_o e^{-\mu x}$$

$$= I_o (1 - e^{-\mu x})$$

$$\cong I_o (\mu x) \text{ for small values of } \mu x.$$

For air, μ for gamma photons of 1.25 Mev energy is $3.2 \times 10^{-5} \text{cm}^{-1}$. (See graph on page 143 of the Radiological Health Handbook). Thus the number of gamma photons absorbed per cubic centimeter at one yard distance from a point source of 1 curie of Co^{60} would be:

$$I_o - I = I_o (\mu x)$$

$$= \left(\frac{7.4 \times 10^{5} \text{ photons}}{\text{cm}^{3}\text{-sec}} \right) \left(\frac{3.2 \times 10^{-5}}{\text{cm}} \right) (1 \text{ cm})$$

$$= 24 \text{ photons/cm}^{3} \text{ - sec.}$$

Each photon was assumed to have an average energy of 1.25 Mev. Thus the amount of energy absorbed would be:

$$(24 \text{ photons/cm}^{3} \text{ - sec}) (1.25 \text{ Mev/photon})$$

$$= 30 \text{ Mev/cm}^{3} \text{ - sec}$$

$$= (30 \text{ Mev/cm}^{3} \text{ - sec}) (60 \text{ sec/min}) (60 \text{ min/hour})$$

$$= 108{,}000 \text{ Mev/cm}^{3} \text{ - hr.}$$

The absorption of 7.03×10^{4} Mev/cm^{3} is equivalent to one roentgen (Page 47 of the Radiological Health Handbook). Thus, converting 108,000 Mev/cm^{3}- hr. to roentgens/hr.

$$= (108{,}000 \text{ Mev/cm}^{3} \text{ - hr}) (1 \text{ roentgen}/7.03 \times 10^{4} \text{ Mev/cm}^{3})$$

$$= 1.54 \text{ roentgens/hour.}$$

Therefore, on the basis of these calculations, it can be concluded that a point source of 1 curie of Co^{60} yields an exposure dose rate of approximately 1.54 roentgens per hour at one yard.

III. SHORT-CUT METHODS

A. General Equations

Several general equations for the exposure dose rate due to gamma point sources have been formulated. These vary as to their accuracy but are very useful for approximations. (See page 56, Radiological Health Handbook).

1. BASED UPON ENERGY AND NUMBER OF GAMMA PHOTONS PER DISINTEGRATION

$$I_\gamma = 0.156\, n\, E\, (10^5 \mu_a)$$

where:

I_γ = mr/hr at 1 meter per millicurie

n = number of gamma photons per disintegration

E = gamma energy in Mev

μ_a = energy absorption coefficient for the particular gamma photons in air in cm^{-1}.

2. BASED UPON NUMBER OF CURIES AND GAMMA ENERGY

$$I = \frac{6CEn}{D^2}$$

where:

I = r/hr at distance D ft.

C = source strength in curies

E = gamma photon energy in Mev

n = number photons per disintegration

D = distance from source in feet.

3. EXAMPLE

Using the approximate equation given in Part A above, the exposure dose rate for a one millicurie source of Na-24 can be estimated.

The "Table of Isotopes" lists Na-24 as emitting two gamma photons per disintegration. These photons possess energies of 1.38 and 2.76 Mev, respectively. The linear absorption coefficient for photons of 1.38 Mev is $0.31 \times 10^{-4} cm^{-1}$; that for photons of 2.76 Mev is $0.265 \times 10^{-4} cm^{-1}$. (Page 145 of the Radiological Health Handbook).

Thus:

$$\begin{aligned} I &= (0.156)\,(n)\,(E)\,(10^5 \mu_a) \\ &= (0.156)\,(1)\,(1.38)\,(0.31 \times 10^{-4})\,(10^5) \\ &+ (0.156)\,(1)\,(2.76)\,(0.265 \times 10^{-4})\,(10^5) \\ &= (0.156)\,(0.43 + 0.73)\,(10) \\ &= 1.8 \text{ mr/hr at 1 meter per millicurie} \end{aligned}$$

B. Experimental Data

Radiation levels at specified distances from several selected radionuclides have been accurately measured.

1. RA-226

The exposure dose rate at 1 yard distance from a one curie source of Ra-226 in equilibrium with its decay products is 1.005 roentgens per hour. (Page 139 of the Radiological Health Handbook).

IV. DISTRIBUTED SOURCES

The data presented relate to point sources only. Where the radioactive material causing the exposure is distributed over an area, or within a volume, modifications of these data will have to be made. (4) (5)

Typical distributed sources include:

A. Linear Sources

For example, "spent" fuel elements, or radioactive materials present in water flowing through a pipe.

B. Sources Distributed Over an Area

For example, surface contamination such as radioactive materials spilled on a laboratory bench or weapons testing fallout deposited on the ground.

C. Sources Distributed Within a Volume

For example, radioactive gases or dusts present in the air; or liquid wastes stored in tanks.

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PRELIMINARY DRAFT

Biological Effects of Ionizing Radiation

I. Introduction

The fact that ionizing radiation produces biological damage has been known for many years. The first case of human injury was reported in the literature just a few months following Roentgen's original paper in 1895 announcing the discovery of X-rays. As early as 1902, the first case of X-ray induced cancer was reported in the literature.

Early human evidence for harmful effects as a result of exposure to radiation in large amounts existed in the 1920's and 30's, based upon the experience of early radiologists, miners exposed to airborne radioactivity underground, persons working in the radium industry, and other special occupational groups. The long term biological significance of smaller, chronic doses of radiation, however, was not widely appreciated until relatively recently, and most of our current knowledge of the biological effects of radiation has been accumulated since World War II.

II. Sequential Pattern of Biological Effects

In general, the sequence of events following radiation exposure may be classified as follows:

A. Latent Period

Following the initial radiation event, and often before the first detectable effect occurs, there is a time lag referred to as the latent period. There is a vast time range possible in the latent period. In fact, the biological effects of radiation are arbitrarily divided into short-term and long-term effects on this basis. Those effects which appear within a matter of minutes, days, or weeks are called short-term effects and those which appear years, decades, and sometimes generations later are called long-term effects.

B. Period of Demonstrable Effects on Cells and Tissues

During or immediately following the latent period, certain discrete effects can be observed.

One of the phenomena that is seen most frequently in growing tissues exposed to radiation is the cessation of mitosis or cell division. This may be temporary or permanent, depending upon the radiation dosage. Other effects observed are chromosome breaks, clumping of chromatin, formation of giant cells or other abnormal mitoses, increased granularity of cytoplasm, changes in staining characteristics, changes in motility or ciliary activity, cytolysis, vacuolization, altered viscosity of protoplasm, and altered permeability of the cell wall.

It should be pointed out that many of these effects can be duplicated individually with other types of stimuli. However, the entire gamut of effects cannot be reproduced by any single chemical agent. (It is interesting to note that a group of chemical agents that come close to duplicating radiation effects are the nitrogen mustards.)

C. Recovery Period

Following exposure to radiation, recovery can and does take place to a certain extent. This is particularly manifest in the case of the short-term effects, i.e., those appearing within a matter of days or weeks after exposure. There is, however, a residual damage from which no recovery occurs, and it is this irreparable injury which can give rise to later long-term effects.

III. Determinants of Biological Effects

A. The Dose-Response Curve

For any biologically harmful agent, it is useful to correlate the dosage administered with the response or damage produced. "Amount of damage" in the case of radiation might be the frequency of a given abnormality in the cells of an irradiated animal, or the incidence of some chronic disease in an irradiated human population. In plotting these two variables, a dose-response curve is produced. With radiation, an important question has been the nature and shape of this curve. Two possibilities are illustrated in figures 1 and 2.

Figure 1 is a typical "threshold" curve. The point at which the curve intersects the abscissa is the threshold dose, i.e., the dose below which there is no response. If a short-term and easily observable radiation effect, such as reddening of the skin, is taken as "response," then this type of curve is applicable. The first evidence of the effect does not occur until a certain minimum dose is reached.

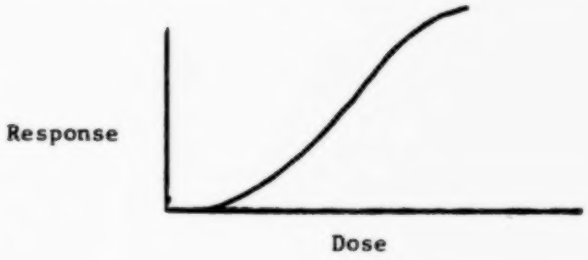

FIGURE 1

Figure 2 represents a linear, or non-threshold relationship, in which the curve intersects the abscissa at the origin. Here any dose, no matter how small, involves some degree of response. There is some evidence that the genetic effects of radiation constitute a non-threshold phenomenon, and one of the underlying (and prudent) assumptions in the establishment of radiation protection guides and in radiation control activities in public health programs has been the existence of a non-threshold effect. Thus some degree of risk is assumed when large populations of people are exposed to even very small amounts of radiation. This assumption often makes the establishment of guidelines for acceptable radiation exposure an enormously complex task, since the concept of "acceptable risk" comes into play, in which the benefit to be accrued from a given radiation exposure must be weighed against its hazard.

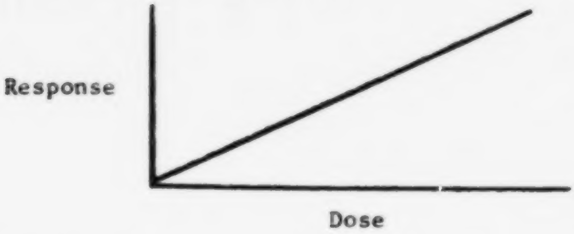

FIGURE 2

B. Rate of Absorption

The rate at which the radiation is administered or absorbed is most important in the determination of what effects will occur. Since a considerable degree of recovery occurs from the radiation damage, a given dose will produce less of an effect if divided (thus allowing time for recovery between dose increments) than if it were given in a single exposure.

C. Area Exposed

Generally when an external radiation exposure is referred to without qualification as to the area of the body involved, whole-body irradiation is assumed. The portion of the body irradiated is an important exposure parameter because the larger the area exposed, other factors being equal, the greater the overall damage to the organism. Even partial shielding of the highly radiosensitive blood-forming organs such as the spleen and bone marrow can mitigate the total effect considerably. An example of this phenemenon is in radiation therapy, in which doses which would be lethal if delivered to the whole body are commonly delivered to very limited areas, e.g. to tumor sites.

D. Variation in Species and Individual Sensitivity

There is a wide variation in the radiosensitivity of various species. Lethal doses for plants and microorganisms, for example, are usually hundreds of times larger than those for mammals. Even among different species of rodents, it is not unusual for one to demonstrate three or four times the sensitivity of another.

Within the same species, biological variability accounts for a difference in sensitivity among individuals. For this reason the lethal dose for each species is expressed in statistical terms, usually as the LD_{50} for that species, or the dose required to kill 50% of the individuals in a large population. For man, the LD_{50} is estimated to be approximately 450 R.

E. Variation in Cell Sensitivity

Within the same individual, a wide variation in succeptibility to radiation damage exists among different types of cells and tissues. In general, those cells which are rapidly dividing or have a potential for rapid division are more sensitive than those which do not divide. Futher, cells which are non-differentiated (i.e., primitive, or non-specialized) are more sensitive than those which are highly specialized. Within the same cell families then, the immature forms, which are generally primitive and rapidly dividing, are more radiosensitive than the older, mature cells which have specialized in function and have ceased to divide.

Based upon these factors, it is possible to rank various kinds of cells in descending order of radiosensitivity. Most sensitive are the white blood cells called lymphocytes, followed by immature forms of other white blood cells found in the bone marrow, and immature red blood cells. Epithelial cells, which line and cover body organs, are of moderately high sensitivity; in terms of injury from large doses of whole-body external radiation, the epithelial cells which line the gastro-intestinal tract are often of particular importance. Cells of low sensitivity include muscle and nerve, which are highly differentiated and do not divide.

IV. Short-term Effects

An acute dose of radiation is one which is delivered to a large portion of the body during a very short time. If the amount of radiation involved is large enough, acute doses may result in effects which can manifest themselves within a period of hours or days. Here the latent period, or time elapsed between the radiation insult and the onset of effects, is relatively short and grows progressively shorter as the dose level is raised. The signs and symptoms which comprise these short-term radiation effects are collectively known as the Acute Radiation Syndrome.

A. Stages in the Acute Radiation Syndrome

1. Prodrome

This is the initial phase of the syndrome, and is usually characterized by nausea, vomiting and malaise. It may be considered analogous to the prodrome stage in acute viral infections, in which the individual is subject to non-specific systemic reactions.

2. Latent Stage

During this phase, which may be likened to the incubation period of a viral infection, the subjective symptoms of illness may subside, and the individual may feel well. Changes, however, may be taking place within the blood-forming organs and elsewhere which will subsequently give rise to the next aspect of the syndrome.

3. Manifest Illness Stage

This phase reflects the clinical picture specifically associated with the radiation injury. Among the possible signs and symptoms are epilation, fever, infection, hemorrhage, severe diarrhea, prostration, disorientation, and cardiovascular collapse. Which, if any, of the foregoing phenomena are observed in a given individual is largely dependent upon the radiation dose received.

4. Recovery or Death

B. Relation of Dose to Type of Acute Radiation Syndrome

Recalling the different sensitivities of various kinds of cells, one can predict roughly the biological systems which will be affected as radiation dose increases. At relatively low doses, for example, the most likely cells to be injured are those with greatest sensitivity, i.e., immature white blood cells of lymph and bone marrow, so that the observable effects during the manifest illness stage would relate to these cells; one would thus expect to observe fever, infection and hemorrhage. This is known as the hematopoietic form of the acute radiation syndrome.

At higher doses, usually over 600 rads, cells of somewhat lower sensitivity will be injured. Of particular importance are the epithelial cells which line the gastrointestinal tract, for when these are destroyed a vital biological barrier is broken down. As a result, there may be fluid loss, overwhelming infection, and severe diarrhea in this gastrointestinal form of the acute radiation syndrome.

In the cerebral form, which may result from doses of 10,000 rads or more, the relatively resistant cells of the central nervous system are damaged, and the affected individual undergoes a rapid illness, characterized by disorientation and shock.

Considering the large degree of individual variation which exists with respect to radiation injury, it is difficult to assign a precise dose range to each of the above forms of the syndrome. The following generalizations, however, may serve to provide a rough indication of the kinds of doses involved. At 50 rads or less, ordinary laboratory or clinical methods will show no indications of injury. At 100 rads, most individuals show no symptoms, although a small percentage may show mild blood changes. At 200 rads, most persons show definite signs of injury; this dose level may prove fatal to those individuals most sensitive to the effects of radiation. At 450 rads, the median lethal dose has been reached, and 50 per cent of exposed individuals will succumb. Approximately 600 rads usually marks the threshold of the gastrointestinal form of the acute radiation syndrome, with a very poor prognosis for all individuals involved; a fatal outcome may well be certain at 800 to 1,000 rads. (For a detailed account of the Acute Radiation Syndrome, including symptomatology, laboratory findings, diagnosis and patient management, see: Thoma, George E., and Wald, Niel, "The Diagnosis and Management of Accidental Radiation Injury," Journal of Occupational Medicine, 1: 421-427, August 1959.)

C. Summary

1. The Acute Radiation Syndrome represents the signs and symptoms which result from large doses of radiation, generally over 100 rads, delivered to a major portion of the body; it is important to recall that this type of injury occurs only when the dose is received over a short period of time. The total effect may vary from mild and transient illness to death.

2. Because of the variation in succeptibility to radiation injury which exists among different individuals, it is extremely difficult to predict with accuracy the degree of effect in a given person, even when the dose is known. Within certain broad ranges, however, certain effects may be correlated with various dose levels on a population basis.

3. The course of the syndrome may vary from hours to several weeks. This time element is generally related to the radiation dose; other factors being equal, the duration of the various phases of the syndrome is inversely proportional to the amount of radiation received, with a rapid, fulminating illness being characteristic of very high doses.

V. Long-term Effects

A. Introduction

Long-term effects of radiation are those which may manifest themselves years after the original exposure. The latent period, then, is much longer than that associated with the acute radiation syndrome. Delayed radiation effects may result from previous acute, high-dose exposures or from chronic low level exposures over a period of years. From the standpoint of public health significance, the possibility of long-term effects on the large number of people receiving low, chronic exposures is cause for greater concern than the short-term radiation effects from acute exposures which involve only a few individuals.

It should be emphasized that there is no unique disease associated with the long-term effects of radiation; these effects manifest themselves in human populations simply as a statistical increase in the incidence of certain already-existing conditions. Because of the low normal incidence of these conditions, it is usually necessary to observe large populations of irradiated persons in order to measure this kind of increase, and employ biostatistical and epidemiologic methodology. In addition to the large numbers of people needed for human studies of long-term radiation effects, the situation is further complicated by the latent period; in some cases, a radiation induced increase in a disease may go unrecorded unless the study is continued for many years.

It should also be noted that although it is possible to perform true experiments with animal populations, in which all factors with the exception of radiation exposure are kept identical in study populations, human data is limited to "second hand" information, accrued from populations which have been irradiated for reasons other than radiobiological information. It is often the special characteristics of irradiated human populations, i.e., the presence of some pre-existing disease, which make it extremely difficult to draw meaningful conclusions when these groups are compared with non-irradiated ones.

Despite the above difficulties, many epidemiologic investigations of irradiated human beings have provided convincing evidence that ionizing radiation may indeed result in an increased risk of certain diseases long after the initial exposure. This information supplements and corroborates that gained from animal experimentation which demonstrates these same effects.

Among the long term effects thus far observed have been somatic damage, which may result in an increased incidence of cancer, embryological defects, cataracts, and lifespan shortening; and genetic mutations, which may adversely affect many generations after the original radiation damage.

B. Carcinogenic Effects

With proper selection of animal species and strains, and of dose, ionizing radiation may be shown to exert an almost universal carcinogenic action, resulting in tumors in a great variety of organs and tissues. There is human evidence as well that radiation may contribute to the induction of various kinds of neoplastic diseases.

1. Possible Mechanisms

It should be made clear that even with high doses of radiation, most irradiated individuals will not suffer long-term consequences despite the fact that the incidence of certain diseases, such as leukemia and other forms of cancer, may be increased many fold. The explanation lies in the fact that most diseases are probably "caused" by the simultaneous interaction of several factors, and that the presence of some of these factors without the others may not be sufficient to induce the disease. Radiation, like other chemical and physical agents which are considered carcinogenic, may be only one of a number of interacting factors which, in a given individual, must be present in order to result in the disease.

Among the tentative explanations thus far proposed for the carcinogenic action of radiation are the following:

a. The essential factor in the production of cancers may be a virus which attacks normal cells by injecting itself into the cell nucleus. The genetic material of the virus, now within the host cell, stimulates the cell to reproduce wildly, thus initiating the neoplastic growth. The virus, it would seem, is ubiquitous, and even may be passed from mother to fetus during prenatal life. It would follow, then, that normal cells have a natural mechanism whereby the action of these viruses is resisted; it is possible that radiation and other carcinogenic agents may activate the latent virus or interfere with the cell's resistance.(1) In this case, the presence of the virus in the tissue would be the primary, initiating event, and the radiation would act as a promoting agent.

b. Certain diseases, among them leukemia, have been associated with specific chromosome aberrations. It may be that radiation damage can produce these abnormalities in the chromosomes and that these changes in turn initiate the disease.(2)

c. Radiation can produce mutations in the somatic cells as well as the germ cells, and it may be that a sufficient accumulation of mutations in a colony of cells can result ultimately in the kind of uncontrolled growth which results in cancer. The somatic mutation concept is an attractive one since it provides a means by which to relate both the carcinogenic effects of radiation and aging. Somatic mutations probably occur constantly at a low rate in all organisms and the resultant damage accumulates gradually in the affected tissues. When the level of malfunction or damage reaches a critical point, cell death or carcinogenesis could occur. Radiation, like other harmful agents, may accelerate the rate at which these mutations occur, thus hastening the death of the organism or the production of cancers.(3)

When radiation doses are large enough to destroy a portion of the cells in an organ, the surviving cells, many of which may have undergone mutations as a result of the radiation exposure, are stimulated to rapid reproduction in order to replace the missing ones. This resulting rapid division may be a concomitant factor in cancer production.(4) In the somatic chromosome abberation or mutation processes, the radiation induced change may be the primary or initiating event, with other factors playing a contributory or promoting role.

d. As a result of the irradiation of water-molecules, which are abundant in all living cells, certain short-lived but potent damaging agents called free radicals are formed and may play an important role in both cancer and aging. There is some evidence that these radicals are generated continually at a low rate as a by-product of certain normal biochemical reactions in living cells, and that radiation simply accelerates their formation.(5)

None of the above speculations need exclude the others. That cancer, mutations and aging are interrelated seems reasonably clear, but the precise mechanisms involved have yet to be elucidated. One theory ties together several phenomena and holds that cancer evolves when a gentic mutation which predisposes an individual to cancer interacts during his lifetime with a sufficiently large number of accumulated somatic mutations (which can be produced by radiation) in the presence of a triggering agent, which could be a virus.(6)

2. Human Evidence

Both empirical observations and epidemiologic studies of irradiated individuals have more or less consistently demonstrated the carcinogenic properties of radiation. Some of these findings are summarized below.

a. Radium Dial Painters. Early in this century, when long-term radiation effects were little recognized, luminous numerals on watches and clocks were painted by hand with fine sable brushes, dipped first in the radium-containing paint and then often tipped on the lips or tongue. Young girls commonly were employed in this occupation. Years later, studies of these individuals who had ingested radium paint have disclosed an increased incidence of bone sarcomas and other malignancies, resulting from the burdens of radium which had accumulated in their bones.(7)

b. Radiologists and dentists. Some early medical and dental users of X-rays, largely unaware of the hazards involved, accumulated considerable doses of radiation. As early as the year 1910, there were reports of cancer deaths among physicians, presumably attributable to X-ray exposure. Skin cancer was a notable finding among these early practitioners;(8) dentists, for example, developed lesions on the fingers with which they repeatedly held dental films in their patients' mouths.

Although the excesses associated with the very early use of X-rays have diminished considerably, recent studies of radiologists have shown a statistically significant increase in the incidence of leukemia as compared with physicians who do not use radiation.(9) (10) A recent similar study with dentists disclosed no significant increase over expected rates.(11)

c. Uranium Miners. Early in this century, certain large mines in Europe were worked for pitchblende, a uranium ore. Lung cancer was highly prevalent among the miners as a result of the inhalation of large quantities of airborne radioactive materials. It was estimated that the risk of lung cancer in the pitchblende miners was at least 50% higher than that of the general population.(7)

Modern mining conditions have greatly improved. Nonetheless, recent studies have indicated a slight but statistically significant excess risk of lung cancer even among contemporary American uranium miners.(12)

d. Atomic Bomb Survivors. One of the strongest supports for the concept that radiation is a leukemogenic agent in man comes from the epidemiologic studies of the survivors of the atomic bombing in Hiroshima. Survivors exposed to radiation above an estimated dose of approximately 100 rem showed a significant increase in the incidence of leukemia. In addition, leukemia incidence correlated well with the estimated dose (expressed as distance from the detonation) thus strengthening the hypothesis that the excess leukemia cases were indeed attributable to the radiation exposure.(7) There is also some indication of an increase in thyroid cancer among the heavily irradiated survivors.(26)

e. Ankylosing Spondylitis Patients. Ankylosing spondylitis, a progressively disabling arthritic disease of the spine, can be treated with large X-ray doses delivered to the vertebrae to slow the progress of the disease and to relieve symptoms. Persons thus treated accumulated large doses of radiation to the bone marrow. A study of a large population of such persons revealed a much higher incidence of leukemia than might be expected in the general population.(7) That the observed leukemias were probably due to the radiation is attested to by a subsequent study in which the leukemia incidence for patients with arthritic diseases not treated with X-rays was ascertained; the authors conclude that the irradiation was probably a major factor in the excess leukemia found among the X-ray treated patients.(13)

f. Children Irradiated for Thymus Enlargement. Many young children with respiratory distress were diagnosed in former years as having enlarged thymus glands and were treated with therapeutic doses of X-rays to the thymic region. A number of follow-up studies were performed on these children, and although results varied, it is generally agreed that these persons have experienced a significantly increased incidence of thyroid cancer and other malignancies of the head and neck.(7) (14)Such findings are not limited to thymic irradiation; further studies (44)have demonstrated excess thyroid cancers and other head and neck malignancies as a result of childhood irradiation to this area of the body for the treatment of a wide variety of benign conditions such as enlarged tonsils and adenoids, acne, etc.

g. Polycythemia Patients. Polycythemia vera is a disease characterized by an excess of red blood cells which can be treated by the administration of phosphorus-32 internally. An epidemiologic study of patients thus irradiated seems to indicate a considerable excess of subsequent leukemias, compared with a control population consisting of non-irradiated polycythemia patients.(15)

h. Patients Treated by Pneumothorax. A link between high doses of radiation to the chest and breast cancer was discovered in a survey of women who had been treated for tuberculosis by artificial pneumothorax,(40)a procedure which consisted of intentionally collapsing the affected lung for a period of time and then re-inflating it. This was accomplished with the assistance of the fluoroscope, and in many cases the pneumothorax treatment was repeated, sometimes more than 100 times. The patient was often positioned in the vertical fluoroscope machine facing the X-ray tube, so that the largest radiation dose was delivered to the anterior surface of the chest. The incidence of breast cancer among these heavily irradiated patients was found to be 4 to 8 times the expected rate for this disease. Further, the investigator was able to show a correlation between the side of the chest receiving the treatment and the affected breast.

i. Children of Mothers Irradiated During Pregnancy. A pioneering study in this area purported to show an increased risk of leukemia among young children if they had been irradiated in utero as a result of pelvic X-ray examination of the mother.(16) Mothers of leukemic children were questioned as to their radiation histories during pregnancy with the child in question, and these responses were compared with those of a control group, consisting of mothers of healthy playmates of the leukemic children. Originally this work received much criticism, based partly on the questionnaire technique used to elicit the information concerning radiation history. It was believed that differences in recall between the two groups of mothers might have biased the results. A larger subsequent study (17) designed to correct for the objections to the first one corroborated its essential findings, and established the leukemogenic effect on the fetus of prenatal X-rays.

It should be noted that the investigations presented thus far which demonstrate the carcinogenic properties of radiation involve large doses, such as those received in therapeutic X-ray procedures, with the exception of these childhood leukemia investigations. Here, doses of radiation are low, in the diagnostic radiographic range. Such findings bear out the high sensitivity of embryonic tissues to radiation damage. In evaluating human studies of the kind described above, two important concepts should be borne in mind.

(1) Because these studies were not designed as radiobiological experiments in which all factors are held constant with the exception of radiation exposure, caution is required before the association between radiation and some later disease can be labeled as a cause-effect relationship. This is particularly true when the study group consists of patients irradiated for some disease or abnormality, since the question arises as to whether the abnormality itself might not account for the later disease rather than the irradiation. Sometimes further studies or the selection of a proper control group with which to compare the irradiated subjects can help to resolve these doubts. For

example, in the spondylitis investigation, a valid question arose as to whether the disease itself might predispose the patient to develop leukemia later. A follow-up study which ascertained the leukemia incidence among non-irradiated rheumatic patients, helped to answer this question. The same question might have arisen in the polycythemia study, except that the control group in this instance consisted of persons with the disease who had not been treated with radiation. Doubts concerning the studies of children with thymic irradiation could have been forestalled, had the control group consisted of children with diagnosed thymic enlargement who had not been treated with X-rays. It was not possible, however, to select such a control group; instead healthy siblings or cancer incidence from general population statistics were employed. Thus, although the weight of evidence in these studies points toward true radiation carcinogenesis, there is room for speculation as to whether the infants with enlarged thymus glands might not have been at least somewhat predisposed to the development of malignant diseases. Even the studies of the relationship of prenatal X-ray examinations to childhood leukemias have been subject to the same kind of question, e.g., whether the special characteristics of the mothers and children in question which necessitated the pelvic X-rays in the first place might not be a predisposing factor in the development of leukemia, irrespective of or in addition to the radiation received. Despite these reservations, all of the above studies, when taken together, comprise an impressive accumulation of evidence indicating that ionizing radiation is a true carcinogen in man.

(2) Assuming that studies such as these are valid, the question arises as to their practical implications. Even in investigations such as the spondylitis and polycythemia studies in which roughly a ten-fold increase in leukemia was observed, the additional risk to an irradiated individual remains small because of the relatively low normal incidence of leukemia. The small but real increase in risk to the individual calls for an intelligent balancing, in each case, of the benefits to be accrued from the radiation exposure and the concomitant hazard. Valuable therapeutic and diagnostic X-ray techniques which are of great benefit to patients cannot be abandoned because of the risk of delayed harmful effects; on the other hand, if the same diagnostic information or therapeutic results can be obtained using techniques which reduce radiation exposure to the patient, or if equally effective non-radiologic procedures which do not involve such risk are available, such methods should be used.

C. Embryological Effects. Considering the fact that immature, undifferentiated, and rapidly dividing cells are highly sensitive to radiation, it is not surprising that embryonic and fetal tissues are readily damaged by relatively low doses of radiation. It has been shown in animal experiments that deleterious effects may be produced with doses of only 10 rads delivered to the embryo. There is no reason to doubt that the human embryo is equally susceptible.(19)

The majority of the anomalies which are produced by prenatal irradiation involve the central nervous system, although the specific type of damage is related to the dose and to the stage of pregnancy during which irradiation takes place. In terms of embryonic death, the very earliest stages of pregnancy, perhaps the first few weeks in the human beings, are most radiosensitive.(19) From the standpoint of practical radiation protection, this very early sensitivity is of great significance, because it involves a stage in human embryonic development in which pregnancy may well be unsuspected. For this reason, the International Committee on Radiological Protection has recommended that routine non-emergency diagnostic irradiation involving the pelvic area of women in the child-bearing years be limited to the 10-day interval following the onset of menstruation. Such precautions would virtually eliminate the possibility of inadvertantly exposing a fertilized egg.(20) This recommendation has been accepted in many countries, including Great Britain,(21) Germany,(22) Norway,(23) and Denmark,(24) among others. In a few countries, guidelines have also been formulated for therapeutic abortion when pelvic radiation doses during certain stages of pregnancy are estimated to exceed specified limits.(22) (24) (25)

For the production of congenital anomalies in the newborn, irradiation during the period of organogenesis is of greatest importance.(19) This period occurs during approximately the second through the sixth week of human gestation, when pregnancy could still be unsuspected. During this period, embryonic death is less likely than in the extremely early stage, but the production of morphological defects in the newborn is a major consideration.

During later stages of pregnancy, embryonic tissue is more resistant to gross and easily observable damage. However, functional changes, particularly those involving the central nervous system, may result from such late exposures(41) and would be difficult to measure or evaluate at birth. They usually involve subtle alterations in such phenomena as learning patterns and development and may have a considerable latent period before they manifest themselves.(17) There is some evidence that the decreasing sensitivity of the fetus to gross radiation damage as pregnancy progresses may not apply for the leukemogenic effects of prenatal irradiation.(16) (28) Another important factor to be considered in evaluating the radiation hazard during late pregnancy is that irradiation may produce true genetic mutations in the immature germ cells of the fetus for which no threshold dose has been established.(19)

It should be emphasized that radiation is not unique in producing embryological effects and that a growing body of evidence exists which indicates that a host of external insults, including certain drugs, chemicals, and viral infections, also can damage the highly sensitive embryo and fetus.

Human evidence for embryological damage has been found among persons exposed in utero at the time of the atomic bombing of Hiroshima. Data obtained from their follow-up through childhood and adolescence into adulthood, in which groups exposed during gestation were compared with each other and with non-exposed controls on the basis of distance from the detonation and stage of pregnancy, shows a growth stunting effect among the exposed, particularly as regards head size, and an increased incidence of mental retardation.

The incidence of these developmental defects correlates well with proximity to the bomb and thus with estimated dose; also, the results further corroborate the increased sensitivity to embryological damage during the first trimester of pregnancy.(29)

D. Cataractogenic Effects. The fibers which comprise the lens of the eye are specialized to transmit light. Damage to these, and particularly to the developing immature cells which give rise to them, can result in opacities in the lens called cataracts, which, if they are large enough, can interfere with vision. Radiation in sufficiently high doses can induce the formation of cataracts; the required dose for humans, which is difficult to ascertain, probably is in the order of several hundred rads for X or gamma rays, and 1/5 to 1/10 of this for neutron irradiation.(7)

Human evidence for radiation cataractogenesis is derived mainly from a relatively small number of workers inadvertently exposed to large doses of radiation to the eye,(7) including several nuclear physicists working with cyclotrons; patients exposed to therapeutic radiation (sometimes from radium plaques applied to the eye);(7) and Japanese atomic bomb survivors who were heavily irradiated.(7) (30)

E. Lifespan Shortening. In a number of animal experimen[illegible]ation has been demonstrated to have a lifespan shortening e[illegible]. The aging process per se is complex and largely obscu[illegible] and the exact mechanisms involved in this effect are as yet uncertain. Irradiated animals in these investigations appear to die of the same diseases as the non-irradiated controls, but they do so at an earlier age. (7) How much of the total effect is due to premature aging and how much to an increased incidence of radiation-induced diseases is still unresolved. (7)

1. Mechanisms. A number of theories have been proposed to account for the phenomenon of aging in general, and for the aging effects of radiation in particular. One theory (7) is that a variety of extrinsic insults produce tissue damage in organisms, some of which is reparable and some of which is irreparable. The irreparable, or residual components of various insults to the organism (infections, trauma, etc.) are additive and cumulative, and when a certain critical amount of injury has accumulated, the organism dies. Because irradiation is one of the agents which can produce such injury, irradiated animals arrive at a lethal accumulation sooner than do the non-irradiated controls. Another theory (31) proposes that radiation exerts its lifespan shortening effect by producing somatic mutations in the cells, which lower the organism's ability to function properly. It further states that organs having cells which seldom, if ever, divide are affected most by these mutations, and play a major part in the aging process.

2. Human Evidence. A study of the possible lifespan shortening effect of radiation in British radiologists yielded negative results. (32) On the other hand, an investigation of American radiologists, as compared with physicians not using radiation, appeared to show that radiologists die approximately 5 years sooner than their non-radiological counterparts. (33) A follow-up study, however, indicated that proper age correction factors had been ignored in the original work, and that when these were applied the lifespan differences disappeared. (34) A more recent American study, (35) in which death rates from various causes were established for radiologists and for two control groups, consisting of physicians who used radiation somewhat and those who did not use it at all, seems to show a true lifespan shortening effect among the radiologists. An association, of course, between two phenomena (in this case, being a radiologist and having a shorter lifespan than a non-radiologist) does not necessarily indicate that one is a cause of the other, and recalling the caution with which epidemiologic studies must be interpreted one might propose explanations for the association other than

that the radiation causes the shortened lifespan. It is conceivable, for example, that those individuals choosing radiology as a specialty could be somewhat less healthy as a group than those selecting other fields, and thus a slightly shorter lifespan among the former would not necessarily relate to their radiation exposure. The authors have considered such alternative explanations of the data, however, and, because of certain strengthening factors in their findings, they nonetheless hold that a true lifespan shortening effect was operative. An important finding in this study, which helps to support the general adequacy of current occupational radiation protection guidelines, is that the excess risk of death among the radiologists is largely confined to those who practiced during the earlier years of X-ray use when safety practices were more lax and occupational exposures among radiologists were presumably much higher than today.

F. Genetic Effects

1. Background

The fertilized egg, which is a single cell resulting from the union of sperm and egg, and which after millions of cell divisions results in a new organism, contains all of the genetic information necessary to produce all of the organs and tissues of the new individual. This information is carried in the nucleus of the fertilized egg on rod shaped structures called chromosomes, arranged in 23 pairs. In each pair, one member is contributed by the mother and the other by the father. With each cell division which the rapidly developing embryonic tissue undergoes, all of this information is faithfully duplicated, so that the nucleus in each cell of the new organism contains essentially all of the information. This, of course, includes those germ cells in the new organism which are destined to become sperm and egg, and thus the information is transmitted from one generation to the next. This hereditary information is often likened to a template, or to a code, which is reproduced millions of times over with remarkable accuracy. It is possible to damage the hereditary material in the cell nucleus by means of external influences, and when this is [illegible] the garbled or distorted genetic information will be reproduced just as faithfully when the cell divides as was the original message. When this kind of alteration occurs in those cells of the ovaries or testes which will become mature sperm and egg, it is referred to as genetic mutation; if the damaged sperm or egg cell is then utilized in conception, the defect is reproduced in all of the cells of the new organism which results from this conception, including those which will become sperm or egg, and thus whatever defect resulted from the original mutation can be passed on for many generations.

Most geneticists agree that the great preponderance of genetic mutations are harmful.(36) By virtue of their damaging effects, they can be gradually eliminated from a population by natural means, since individuals afflicted with this damage are less likely to reproduce themselves successfully than normal individuals. The more severe the deficit produced by a given mutation, the more rapidly it will be eliminated, and vice-versa; mildly damaging mutations may require a great many generations before they gradually disappear.(37)

As a balance to this natural elimination of harmful mutations, fresh ones are constantly occurring. A large number of agents have mutagenic properties, and it is probably that our current knowledge includes just a fraction of these.(38) In addition, it may be that mutations can arise within the germ cells of an organism without external insult; free radicals, which may be produced as a natural by-product of normal metabolic reactions in the body, may have a mutational effect.(5) Among the various external influences which have been found to be mutagenic are a wide variety of chemicals, certain drugs, and physical factors such as elevated temperatures and ionizing radiation. Natural background radiation probably accounts for a small proportion of naturally occurring mutations. For man, it has been estimated that background radiation probably produces less than ten per cent of these.(39)Man-made radiation, of course, if delivered to the gonads, can also produce mutations, over and above those which occur spontaneously. Radiation, it should be noted, is not unique in this respect, and is probably one of a number of man-made environmental influences which is capable of increasing the mutation rate.

2. Observing Mutations

Measuring changes in the normal mutation rate in humans is extremely difficult for several reasons. First, the majority of mutations are recessive, that is, their full effects do not manifest themselves in an individual unless he carries the same mutational defect in the same location on a given pair of chromosomes, i.e., unless both his mother and father were afflicted with the same kind of genetic damage.(7) It can be seen from this that it might take many generations after a genetically damaging event occurred in a population before enough individuals carrying the recessive mutation mated with one another to produce offspring who would demonstrate overt damage. Secondly, contrary to popular impression, the damage produced by most mutations is subtle in its effects and difficult to measure. Mutations, for example, may result in a slightly altered metabolism, in which there is a less efficient utilization of certain nutritional elements, or a slightly greater predisposition to a given disease, or a slightly lower intelligence than would otherwise have been attained.(7) Adding to this is the difficulty encountered in extricating true genetic phenomena from other influences which may produce the same results. It would be most difficult, for example, to determine whether an individual's predisposition to heart disease were due to a subtle genetic defect in his cardiovascular system, or to environmental stresses such as diet or occupation, or to personality and developmental factors in childhood, etc.

Despite these difficulties, it is possible to observe fluctuations in the mutation rate if large enough populations are available for close study. Certain diseases, for example, have been linked to specific genetic defects, and an increase or decrease in the incidence of these diseases would indicate a concomitant change in the mutation rate. Certain mutations are lethal; they are highly damaging and result in intrauterine death. Other factors being equal, significant fluctuations in the incidence of these deaths in a population might serve as a rough barometer of changing mutation rates. Observing a population for evidence of genetic mutations can be likened to watching an iceberg; most of the iceberg is invisible, with only a small portion above the water. Changes in the size of the observable part serve to give some indication of the more significant changes taking place beneath the surface. (For a discussion of the difficulties and techniques in finding and evaluating genetic radiation damage, see Chapter 7, "The Quantitative Assessment of Hereditary Damage Induced by Radiation," by James and Newcombe, in "Progress in Medical Genetics," Vol. 3, Grune and Stratton, Inc. New York, 1964.)

Even in the first generation after a population has been exposed to a possibly mutagenic event, it is possible to observe the effects of certain kinds of mutations. Dominant mutations, i.e., those which manifest their full effects even when only one parent carries the mutation, may be evident in the offspring. Those which are also lethal will appear as an increase in intra-uterine deaths. There is a particular kind of recessive mutation which can also be observed in the first generation, the sex-linked mutation. Of the 23 pairs of chromosomes in the fertilized egg, one pair determines the sex of the offspring. In this pair, females carry two full-sized chromosomes, called X-chromosomes, while males carry one X chromosome and one Y chromosome, which is much smaller and which probably does not carry a full compliment of genetic produced by most mutations is subtle in its effects and difficult to measure. Mutations, for example, may result in a slightly altered metabolism, in which there is a less efficient utilization of certain nutritional elements, or a slightly greater predisposition to a given disease, or a slightly lower intelligence than would otherwise have been attained.(7) Adding to this is the difficulty encountered in extricating true genetic phenomena from other influences which may produce the same results. It would be most difficult, for example, to determine whether an individual's predisposition to heart disease were due to a subtle genetic defect in his cardiovascular system, or to environmental stresses such as diet or occupation, or to personality and developmental factors in childhood, etc.

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3. Animal Evidence

The mutagenic properties of ionizing radiation were first discovered in 1927, using the fruit-fly as the experimental animal. Since that time, experiments have been extended to include other species, and a great deal of recent investigation has been carried out on the mouse. Animal experimentation remains our chief source of information concerning the genetic effects of radiation, and as a result of the intensive experimentation which has been carried out during recent years, certain generalizations may be made.(7) Among those of health significance are (1) that there is no indication of a threshold dose for the genetic effects of radiation, i.e., a dose below which genetic damage does not occur, and (2) that the degree of mutational damage which results from radiation exposure seems to be dose-rate dependent, so that a given dose is less effective in producing damage if it is protracted or fractionated over a long period of time.

4. Human Evidence

a. A major human study on genetic effects has concerned the Japanese atomic bomb survivors. As the index of a possible increase in the mutation rate, the sex-ratio in the offspring of certain irradiated groups (families, for example, in which the mother had been irradiated but the father had not) was observed, using the approach described earlier. Assuming that some of the mutational damage in the mothers would be recessive, lethal and sex-linked, a shift in the sex-ratio among these families might be expected in the direction of fewer male births than in completely non-irradiated groups, and this seemed to be the case in early reports.(7) Later evaluation of more complete data, however, did not bear out the original suggestion of an effect on the sex-ratio.(43)

b. The pre-conception radiation histories of the parents of leukemic children as compared with those of normal children was a part of the subject of another investigation.(28) From the results, it would appear that there is a statistically significant increase in leukemia risk among children whose mothers had received diagnostic X-rays during this period. The effect here is apparently a genetic rather than an embryologic one, since the irradiation occurred prior to the conception of the child.

A somewhat similar study ascertained the radiation exposure histories of the parents of children with Mongolism; most of this exposure too, was prior to the conception of the child.(42) A significantly greater number of the mothers of Mongoloid children reported receiving diagnostic fluoroscopy and X-ray therapy prior to the birth of the Mongoloid child than did mothers of normal children comprising a control group.

The findings of these two studies serve to provide additional evidence that ionizing radiation is a mutational agent in man. On the other hand, they can be viewed with the same kind of reservations as were explained previously, i.e., there could be significantly differences to begin with between populations of people requiring X-rays and those not requiring X-rays; these differences alone might account for a slightly higher incidence of leukemia or Mongolism in the offspring of the former group, irrespective of the radiation received.

5. Health Significance of Genetic Mutations

Recalling the previous discussion concerning the natural elimination of harmful mutations and the simultaneous introduction of fresh ones into a population, the total number of mutations present may be likened to the water in a tank, in which in-flow at the top represents new mutations and out-flow at the bottom represents the eliminations of old mutations. The water level does not necessarily remain constant; if the rate at which new mutations are produced exceeds that at which old ones are discarded, the pool of mutations grows larger. The reverse is true if the output exceeds the input with a resultant lowering of the pool. With contemporary human populations, it is highly desirable to keep the level of the mutational pool as low as possible, since the pool largely represents diseases and defects which tend to lower overall biological fitness. However, two factors unique to modern life may tend to increase the level. First, human populations are being exposed to a greater and greater number of potential mutagens as a result of a progressive increase in the variety and quantity of man-made chemical and physical agents which are a product of our technological advances. Secondly, modern medical knowledge and techniques result in the salvage of more and more individuals with genetic defects so that they may reproduce themselves, thus distributing these defects to an ever larger number of people. In the face of these factors which tend to increase the mutational load in the world's population, it is all the more important, if the level of the pool is to be kept at a minimum, to make every effort to maintain the influx of new mutations as small as possible. Considering the potent mutagenic properties of ionizing radiation, the goal, then, is clear: to avoid any unnecessary irradiation of the gonads.

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Basic Principles of Radiation Protection

I. INTRODUCTION

Application of the basic principles for protecting personnel from ionizing radiation will vary quantitatively, depending upon type, intensity, and energy of the radiation source. For example, beta particles from radioactive materials require less shielding than high speed electrons from a high-energy particle accelerator. Ideally, application of these protection principles would reduce radiation exposure to zero, but usually cost and convenience make a compromise necessary. The maximum permissible exposure levels serve as upper limits of this compromise.

II. EXTERNAL RADIATION HAZARDS

A. X Rays and Gamma Rays

X rays and gamma rays constitute the most common external radiation hazards. When they are of sufficient energy, both can penetrate deep into the body. No radiosensitive organ is beyond the range of their damaging powers. The most common source of x rays is, of course, the x-ray machine. Some x rays are also generated as a by-product of certain atomic and nuclear reactions. Gamma-ray sources include nuclear reactors, particle accelerators, and radionuclides. X-ray machines can usually be treated as point sources; this simplifies protective measures somewhat. Gamma-ray sources may or may not be point sources.

B. Beta Particles

Beta particles may constitute an external hazard if they have enough energy to penetrate to the basal layer of the epidermis. Those particles which are stopped by the horny layer of the skin are not external hazards. Radioactive nuclides and high-energy particle accelerators may be sources of beta radiation.

C. Neutrons

Neutrons, because of their penetrating powers, are considered external radiation hazards. They are produced in abundant quantities by high-energy particle accelerators and nuclear reactors. Small sources--such as radium-beryllium capsules--also produce neutrons. Neutrons are perhaps the most dangerous of all external radiation because they can cause much tissue damage and have so far proved to be the most difficult to monitor.

D. Protons and Deuterons

High-energy protons and deuterons are dangerous external radiation hazards due to their high ionizing powers. Fortunately, high-energy particle accelerators which produce them are few in number.

III. CONTROL OF EXTERNAL RADIATION HAZARDS

A. Distance

Distance is a very effective, and in many instances the most easily applied, method of radiation protection. Beta particles of a single energy have a finite range in air. Sometimes the distance afforded by the use of remote control handling devices will give complete protection.

The inverse square law for reduction of radiation intensity applies to point sources of x, gamma, and neutron radiation. (X-ray tubes act enough like point sources for reduction calculations by this law to hold valid. Gamma ray sources whose dimensions are small compared to the distances involved may also be considered point sources, as may capsule neutron sources.) The inverse square law states that radiation intensity from a point varies inversely as the square of the distance from the source. This can be expressed mathematically as: $I_1 / I_2 = (R_2)^2 / (R_1)^2$.

Here I_1 is the radiation intensity at distance R_1 from the source and I_2 is the radiation intensity at distance R_2 from the source. This formula shows that doubling the distance from the source decreases intensity by a factor of four; increasing the distance by a factor of three reduces the radiation intensity to one-ninth of its value, etc. The inverse square law does not apply to extended sources or radiation fields due to multiple sources.

B. Shielding

Shielding is one of the most important methods for radiation protection. To apply shielding methods, one must understand how x and gamma radiation is attenuated in an absorbing medium. Energy is lost by three methods: photoelectric effect, Compton effect, and pair production. The predominating mechanism depends upon the energy of the radiation and the absorbing material. The photoelectric effect is most important at low energies, the Compton effect at intermediate energies, and pair production at high energies. The last cannot occur unless the incident radiation has more than 1.02 MeV of energy. As x- and gamma-ray photons travel through an absorber, their decrease in number caused by these absorption processes is governed by the energy of radiation, the specific absorber medium, and the thickness of the absorber transversed. Mathematically, this may be expressed as:

$$\frac{\Delta I}{I_0} = - \mu \Delta x$$

where:

ΔI = reduction of radiation intensity

I_0 = incident radiation intensity

μ = proportionality constant

Δx = thickness of absorber transversed

This equation can be solved by integration:

$I = I_0 e^{-\mu x}$.

Here e is the natural logarithm base and I is the radiation intensity after absorption by material of thickness x. Using this formula, it is easy to calculate the radiation intensity behind a shield of thickness x, or to calculate the absorber thickness necessary to reduce radiation intensity to a desired level, if the factor μ is known. This factor is called the "linear absorption coefficient" when x is a "linear dimension." However, the value of μ depends upon the energy of the radiation and the absorbing medium. Tables and graphs in the Radiological Health Handbook give values of mass attenuation coefficients determined experimentally for all radiation energies and for many absorbing materials. To obtain μ, multiply the mass attenuation coefficient of the shielding material by its density. The larger the value of μ the greater the reduction in intensity for a given thickness of material. The fact that lead has a high μ for x and gamma radiation is partially responsible for its wide use as a shielding material. Figure 1 shows the relative shielding properties of some common materials.

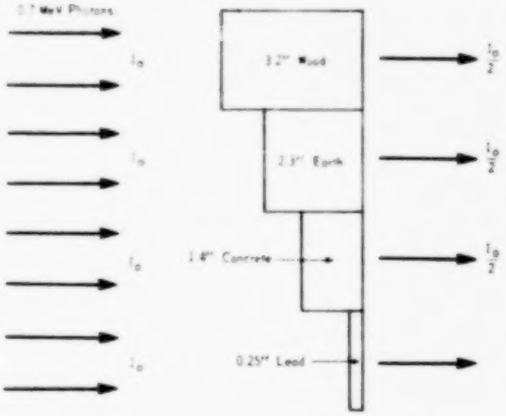

Figure 1.--Half Value Layers for Shielding with some Common Materials

If the radiation being attenuated does not meet narrow beam conditions, and/or thick absorbers are involved, the absorption equation becomes: $I = bI_0e^{-\mu x}$.

Here b is the buildup factor and takes into account increasing radiation intensity from scattered radiation within the absorber. Graphs of buildup factors can be found in the Radiological Health Handbook.

In discussing shielding, there are a few facts to keep in mind:

1. Persons outside the "shadow" cast by the shield are not necessarily protected.
2. A wall or partition is not necessarily a safe shield for persons on the other side.
3. In effect, radiation can "bounce around corners"; i.e., it can be scattered.

The third fact is so important that it merits further discussion. Scattered radiation is present to some extent whenever an absorbing medium is in the path of radiation. The absorber acts as a new source of radiation. Frequently, room walls, floor, and other solid objects are near enough to a source of radiation to make scatter appreciable. When a point source is used under these conditions, the inverse square law is no longer completely valid for computing radiation intensity at a distance. Measurement of the radiation is then necessary to determine the potential exposure at any point.

Shielding will attenuate beta radiation, and it takes relatively little shielding to absorb it completely. Therefore, the general practice is to use enough shielding for complete absorption. For low-energy beta emitters in solution, the glass container generally gives complete protection. In many cases, plastic shielding is effective and convenient. The absorption of higher-energy beta radiation may result in the production of electromagnetic radiation called bremsstrahlung, literally "breaking radiation." This is produced by deceleration of the beta particles when they interact with the shielding material. Electromagnetic radiation is more penetrating than the beta radiation that produced it and must be considered in shielding calculations. Principles used for attenuating x and gamma radiation from conventional sources should be used for shielding against this additional radiation. Since bremsstrahlung is most readily produced when high-energy beta particles interact with high atomic number materials (e.g. lead), this is a compelling reason for the use of glass, plastic, or other low atomic number materials for shielding against beta radiation.

Tables or graphs in the Radiological Health Handbook list the maximum range of beta particles of various energies in different absorbing media. These can be used to calculate the necessary shielding.

Fast neutrons are poorly absorbed by most materials; therefore, they must be slowed down for efficient absorption. Since the greatest transfer of energy takes place in collisions between particles of equal mass, hydrogenous materials are most effective for slowing down fast neutrons. Water, paraffin, and concrete are all rich in hydrogen, and thus important in neutron shielding. Once the neutrons have been reduced in energy, they may be absorbed by either boron or cadmium. When a boron atom captures a neutron, it emits an alpha particle. But because of the extremely short range of alpha particles, there is no additional hazard. Neutron capture by cadmium results in the emission of gamma radiation. Lead or a similar gamma absorber must be used as a shield against these gammas. A complete shield for a capsule-type neutron source may consist of: (1) a thick layer of paraffin to slow down the neutrons, (2) a surrounding layer of cadmium to absorb the slow neutrons, and (3) an outer layer of lead to absorb both the gammas produced in the cadmium and those emanating from the capsule.

C. Exposure Time

It may occasionally be necessary for a radiation worker to perform in areas of high exposure rates. This can be accomplished within the recommendations of the Federal Radiation Council or the National Council on Radiation Protection and Measurements if the total accumulated whole-body dose, within a period of 13 weeks, does not exceed 3 rems and if the average overall accumulated dose does not exceed 5(N−18) rems. To stay within this limit, it may be necessary to work men in relays so that the RPG or MPD is not exceeded by any one man.

IV. INTERNAL RADIATION HAZARDS

A. Modes of Exposure

Radioactive materials may enter the body by ingestion, by inhalation of air containing radioactive materials, by absorption of a solution of radioactive materials through the skin, or by absorption of radioactive material (into the blood stream) through a cut or break in the skin. The danger of ingestion of radioactive materials is not necessarily that of a large amount swallowed at one time, but rather the accumulation of small amounts on hands, cigarettes, food, or other objects, thus bringing the material into the mouth frequently.

B. Sources

Any radioactive material that enters the body is an internal hazard. The extent of the hazard depends upon the type of radiation emitted and its energy, the physical and biological half-life of the material, and the radiosensitivity of the organ where the isotope localizes. Alpha and beta emitters present the most dangerous internal hazard because their specific ionization is very high. (The number of ion pairs formed per centimeter of path in any given medium is called the specific ionization for that particular ionizing radiation.) Polonium is an example of a potentially very serious internal hazard. It emits a highly ionizing alpha particle of 5.3 MeV energy, has a half-life of 138 days, and "creeps" out of containers. Tables in the National Bureau of Standards Handbook 69 list the physical and biological information necessary to evaluate the hazard of most radionuclides.

V. CONTROL OF INTERNAL RADIATION HAZARDS

Contamination prevention and control are the most effective ways to reduce internal hazards. Protective devices and good handling techniques are helpful in this respect. Dust should be kept to a minimum by the elimination of dry sweeping and the use of air filters. Laboratory operations with radioactive materials should be performed in hoods designed to keep room air contamination at a minimum. The exhaust air must be filtered and, if necessary, washed to eliminate any possible public hazard. Protective clothing should be worn so permanent clothing does not become contaminated. These all help limit the spread of contamination.

To prevent inhalation of radioactive materials, respirators should be available for emergency use in areas where the concentration of airborne activity is above maximum permissible levels. Eating and smoking in areas where radioactive materials are handled should be prohibited to reduce the ingestion hazard. Laboratory design and construction should be such that if decontamination becomes necessary, it is easily accomplished. It may be desirable to coat floor, walls, ceilings, and furniture with a strippable material for easy removal and disposal of contamination.

The proper instruments for handling radionuclides should be used to reduce the probability of accidents. There are many more fine points which should be considered for safe handling of radionuclides. Most are discussed in books and booklets [2-4] concerning this topic. Thorough study of such publications should be made before handling radionuclides. Proper planning of operations will eliminate many sources of hazards.

VI. SUMMARY

Time, distance, and shielding are the basic methods for radiation protection against external sources. The application of some combination of these methods will reduce radiation exposure to an acceptable level. The prevention and control of contamination, the use of protective devices, and good techniques should keep internal hazards below the maximum permissible levels.

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Control of Internal Radiation Hazards

I. INTRODUCTION

Internal exposure results from the deposition of radioactive material within the body through inhalation, ingestion, or absorption. Internally deposited radioactive material is extremely hazardous since it produces continuous radiation exposure until it decays or is removed from the body by excretion. Therefore, internal radiation exposure can be controlled only by preventing the entry of radioactive material into the body. It follows that internal exposure control is essentially a problem of contamination control.

II. POTENTIAL INTERNAL RADIATION HAZARDS

The hazard created by a radionuclide inside the body depends on (1) the amount of radionuclide in the organ, (2) the energy of the emitted radiation, (3) the relative biological effectiveness (RBE) of the radiation, (4) the uniformity of distribution within the critical organ, (5) size and essentiality of the organ, and (6) the effective half-life. The effective half-life of a radionuclide is a measure of the decrease in radioactivity in the tissue with time and is determined by combining the radiological half-life, T_r, (radioactivity decay) and biological half-life (T_b body elimination):

$$T_{eff} = \frac{T_b \; T_r}{T_b + T_r}$$

National Bureau of Standards Handbooks 48 and 42 (page 4) group selected radionuclides according to their relative radiotoxicity by weighing the previously mentioned factors and by considering the usual quantities involved and the modes of handling in typical experiments.

A. Alpha Emitters

1. HAZARDS OF ALPHA EMITTERS

Because of their relatively large mass and double positive charge, alpha particles lose their energy very rapidly in the medium through which they pass. The range of the alpha particles is therefore very short. A 4-Mev alpha particle has a range in tissue of only a few hundredths of a millimeter and can travel only 3 centimeters in air. Accordingly, alpha particles do not present an external radiation hazard. However, alpha emitters do present one of the greatest internal hazards. When deposited throughout a vital organ, they can cause great damage because of their relatively high energies (4 to 9 Mev) and high specific ionization, and because their relative damage (RBE) factor to tissue is about 20 times greater than that for beta and gamma radiation.

2. TYPICAL HAZARDOUS ALPHA EMITTING NUCLIDES

Among the alpha emitting nuclides in common use are plutonium-239, polonium-210, natural and enriched uranium, and radium-226. Plutonium-239 emits alpha particles with an average energy of 5.15 Mev and which have a range of about 40 microns in water and soft tissue. It also emits several weak x-rays with energies from 10 to 22 kev. Pu^{239} deposits predominantly in the skeleton where, many years later, it may produce bone disease (including cancer). Its absorption from the gastrointestinal tract is only about 0.003 percent. Lung absorption is potentially the most important means of entry into the body, and may reach 1 to 10 percent of the inhaled dose depending on such parameters as particle size and solubility. The permissible body burden of Pu^{239} is only 0.04 μc. This is an extremely small amount of material equivalent to only 0.625 micrograms or approximately 8.75×10^4 alpha disintegrations per minute. Once fixed in the body, Pu^{239} is excreted very slowly (about 200 years would be required to eliminate one-half the body burden). Polonium-210 is another example of a potential internal hazard. It emits highly ionizing alpha particles of 5.3 Mev energy, has a radiological half-life of 138 days, and "creeps" out of containers. The critical organs are the spleen and kidneys and the effective half-lives are 42 and 46 days respectively. Polonium is often used in neutron sources.

Natural uranium emits 4.4 Mev alpha particles. The critical organs are the kidneys, where soluble uranium exhibits an effective half-life of 15 days. The permissible body burden for the kidney is 5×10^{-3} μc, a relatively large amount of natural uranium (about 40 milligrams). It is unlikely that a person would get this much material in his body, and if he did, it would probably be chemically toxic before the effects of radiation would become apparent. Enriched uranium is a more serious radiological hazard than natural uranium.

Radium-226 and its daughter products emit alpha particles with energies of 4.8 Mev. The effective half-life in bone is approximately 44 years. This radionuclide is a bone seeker and presents a very serious internal radiation hazard. The permissible body burden for bone is 0.1 μc (or 0.1 microgram). Radium in equilibrium with its daughters is commonly used as a gamma source.

B. Beta Emitters

1. HAZARDS OF BETA EMITTERS

Beta particles are lighter and have less charge than alpha particles. This results in a longer path of travel, with less specific ionization.

Accordingly, the relative biological effect of beta particles is much less than that caused by alpha radiation. For example, the absorbed energy from beta radiation must be approximately 20 times greater (per unit mass of tissue) than that of alpha radiation to produce the same damaging effect. Both alpha and beta emitting materials are considered to be primarily internal radiation hazards. However, beta radiation may also represent an external hazard when the energy is sufficient to penetrate the dead layer of skin. External beta burns can be extremely painful and slow to heal.

2. TYPICAL BETA EMITTING ISOTOPES

Strontium-90, a fission product, is a pure beta emitter which decays with a radiological half-life of 27.4 years. Its radioactive daughter, Yttrium-90, is also a pure beta emitter. Strontium-90 has an effective half-life in the bone of approximately 17.5 years. It is a bone seeker and is considered to be a very dangerous internal radiation source.

Iodine-131 is also a fission product and a beta-gamma emitter. It ejects beta particles with average energies of 0.22 Mev. The effective half-life is 7.6 days. The critical organ for irradiation is the thyroid. National Bureau Standards Handbook No. 48 classifies this radioisotope as "moderately dangerous". Because of its short half-life, it would have to be ingested or inhaled in relatively large quantities, or over a relatively long period to be a serious hazard.

III. MODES OF ENTRY

A. Inhalation

Inhalation of radioactive aerosols is one of the principal potential modes of entry into the body. The absorption, retention, and elimination of material taken into the lungs depend on such parameters as particle size of the inhaled material, solubility, and rate of respiration of the individual, and thus pose a very complex problem.

B. Ingestion

The percent of absorption into the blood stream of an orally administered radionuclide depends on its chemical and physical form. A large proportion of ingested material is rapidly excreted in the feces, but, because of irradiation of the intestine and because even a small amount entering the blood stream is highly toxic, ingested radioactive material is very dangerous.

C. Through Skin

Radioactive materials can also enter the blood stream by absorption through the unbroken skin and through abrasions, cuts, and punctures. Thus all personnel working with radionuclides should employ proper procedures and wear protective clothing to prevent contact with the skin.

IV. FUNDAMENTAL PRINCIPLES OF CONTAMINATION CONTROL

Internal radiation exposure can be controlled only by preventing the entry of radioactive material into the body. As previously stated, internal exposure control is essentially a problem of contamination control. The two fundamental principles of contamination control are containment and cleanliness.

A. Containment

The spread of radioactive contamination may be minimized at several points. Containment (i.e. the restriction of active materials to specified areas) is the first line of defense. This is achieved by the adoption of operating techniques and laboratory methods best suited to radionuclides, careful choice of equipment, and proper discipline.

1. OPERATING TECHNIQUES

The choice of processes and techniques for handling radioactive materials is probably the most important part of contamination control. Important aspects to be considered are:

a. Quantity of radioactive material used.--The quantity of radioactive material used should always be as small as is practicable and materials of low toxicity should be chosen whenever possible.

b. Protection of working surfaces.--During radiochemical operations, the working surface should be covered by either metal trays or a layer of disposable absorbent material to soak up spilled liquids. Blotting paper or "diaper" paper (heavy absorbent paper backed with impervious material such as oiled kraft paper) are used extensively. The latter is preferable, as it prevents liquids reaching the bench top.

c. "Dummy runs".--It is worthwhile to carry out a complete "dummy" run with inactive or labeled materials before any manipulation of larger open radioactive sources is undertaken. In this way unexpected difficulties may be discovered, weaknesses in equipment detected, and the procedure modified accordingly.

d. Uses of Containers.--Reliance should never be placed on single containers; suitable drip trays or double containers should be used as a matter of course and they should be capable of holding the entire contents of the primary containers.

e. Handling equipment.--Potentially contaminated equipment should never be handled with bare hands and pipettes must never be operated by mouth suction. Glass blowing in active areas or on contaminated equipment should be done with special techniques which avoid blowing by mouth.

2. EQUIPMENT

a. Radiochemical hoods.--Laboratory operations accompanied by the evolution of radioactive fumes or gases require the use of radiochemical hoods. The inner surfaces of the hoods should be of non-porous material (e.g. stainless steel). The surfaces may also be covered with a strippable paint to permit easy decontamination.

b. Glove boxes.--The glove box is a totally enclosed chamber in which alpha-emitters and low-energy beta sources can be handled safely. The great advantage of glove boxes, apart from the obvious one of confining contamination within a totally enclosed space, is the small air supply involved in comparison with the generous air flow necessary in the usual hood system to prevent escape of activity into the working atmosphere. The disadvantages of glove boxes are the extra time and labor involved and the difficulties of working inside them, but they are indispensable for work with dry powders of highly toxic material above certain levels.

c. Segregation.--All equipment used for handling the active material must be segregated and used only for this work. Marking the "radioactive" equipment helps. This segregation is particularly important for glassware and handling equipment such as tongs and gloves which can very easily transfer contamination from fume cupboards to the open laboratory.

3. DISCIPLINE

a. Clothing.--All personnel working in radiation areas must wear the prescribed protective clothing. The degree of clothing change considered necessary depends on the level of activity and the types of operations. In tracer laboratories, a standard laboratory coat should be worn and a compulsory shoe change may be desirable. In intermediate level laboratories, laboratory coats and a change of shoes should be regarded as compulsory and it may be necessary to have a complete change of clothing plus a shoe change. Associated facilities will therefore range from merely coat-hooks in the laboratory to a suitably located change room.

Individuals working in more hazardous regions (e.g. in the area of hot cells) need a greater degree of protection. They should not work in an active area without the appropriate change of clothing and this clothing should not be worn outside the area. If it is necessary for protective clothing to be worn outside a potentially highly-contaminated laboratory, a surrounding control, or buffer, area should be established and treated as an intermediate level, active area. Gloves should be worn when handling active materials to avoid hand contamination. An effective type of rubber glove is the standard tight-fitting surgical glove made of thin latex. The surgical technique should be employed when putting on and taking off gloves, so as not to contaminate the inside surface.

b. Personal effects.--Unessential personal items should not be taken into the active area. Food, drink, or smoking materials should not be taken into an active area and nothing should be placed in the mouth.

B. Cleanliness

The use of proper operating techniques must be supplemented by cleanliness and good housekeeping. These will minimize the spread of contamination and will prevent the buildup of serious levels.

C. Monitoring

Cleanliness in a radioisotope laboratory goes beyond just visible cleanliness. It involves the regular monitoring of the laboratory equipment, personnel and operating techniques, followed by prompt decontamination where necessary. Working surfaces, floors, and all other surfaces in the area should be checked regularly. The monitoring should be extended to cover all protective clothing. All people and equipment leaving the active area should be monitored for surface contamination.

D. Personnel

The mental attitude of the worker plays an extremely important part in radiation protection. Everyone working in an active area should wash and monitor his hands before leaving, particularly at the end of the day and before eating. Gloves should be monitored regularly and private clothing may also need to be checked.

E. Cleaning

Dust-raising methods such as dry sweeping should be avoided. A separate set of cleaning equipment should be kept solely for use in the active area.

V. MANAGEMENT OF SPILLS

In the event of a "spill" or accident involving the contamination of an area, personnel protection and the immediate confinement of the contamination is of primary importance. When a spill occurs a standardized approach should be followed:

A. Prevent Spread of Contamination

1. Close doors and windows.

2. Turn off fans, air conditioners and other such ventilation if possible.

3. Close ventilation ducts if possible.

4. Get out of room leaving shoes and outer garments at door.

5. Lock door(s) and seal cracks with tape.

The spill is now confined and clean-up can be done later according to a well-considered plan.

B. Decontamination

Work out a specific plan considering the physical facilities and the material spilled.

1. Monitor spill area to determine the extent and hazard of contamination.

2. Proceed with decontamination.

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Radiation Protection Guides

I. PHILOSOPHY OF RADIATION PROTECTION

The setting and execution of guidelines for radiation protection are based upon an underlying philosophy in which two factors are of prime importance. First is the assumption that radiation effects follow a linear, or non-threshold dose-response relationship. There is convincing evidence, particularly insofar as the genetic effects of radiation are concerned, that there exists a non-threshold phenomenon; and although positive proof is thus far lacking, it has been deemed prudent to adopt this more conservative hypothesis in setting protection standards for large numbers of people. According to the non-threshold concept, there is no radiation dose so small that it does not involve some degree of risk. The non-threshold relationship, therefore, implies that there is no radiation protection guideline, no matter how low, which can insure absolute safety to every individual in a large population receiving the guideline dosage. However, since the magnitude of the risk is proportional to the dose received, untoward effects would become manifest at very low dose levels only if extremely large numbers of exposed individuals were observed.

The second major factor to consider is that radiation, like many other developments of modern life (such as the automobile) confers great benefit upon both society and the individual along with its risk to health. Consideration of the extent of these benefits makes a certain degree of risk acceptable. Thus, a balance must be struck in each contemplated radiation usage, in which the benefit to be gained is weighed against the anticipated risk. If the benefit outweighs the risk, the radiation is utilized so that its maximum benefit will be realized while human exposure will be maintained at the minimum consistent with deriving these benefits. The overall public health philosophy, then, is to attain maximum advantage from the use of ionizing radiation while minimizing concomitant exposure, that is, eliminating wherever possible all unnecessary exposure to radiation.

The Radiation Protection Guide (RPG) may be defined as the radiation dose which should not be exceeded without careful consideration of the reasons for so doing. In light of the non-threshold phenomenon, every effort should be made to encourage the maintenance of radiation exposures as far below the guide as practicable. Methods of estimating guides are primarily based on past experience in which individuals have been exposed to radiation for various reasons and upon animal experiments which have contributed greatly to the study of the effects of radiation. From this combined knowledge and from an understanding of the relative biological damage produced by various types of radiation, protection guides for whole-body exposure and for various organs have been recommended. These guides, of course, represent doses far below those at which any effects have thus far been observed.

II. BASIS FOR RADIATION PROTECTION GUIDES

Establishment of "safe" levels of long-term radiation dose requires knowledge of the cause-effect relationship between radiation dose and biological damage. Such damage may appear many years after initial exposure and is usually indistinguishable from the normal diseases and impairments of man. Information accumulated on this subject is, therefore, difficult to evaluate and is often controversial. Nevertheless, observations involving man and animal life have resulted in the accumulation of significant data. These include:

A. Genetic Effects

1. Studies involving fruit flies and mice
2. Hiroshima and Nagasaki populations (inconclusive)

B. Incidence of Neoplasms

1. Bone tumors in radium dial painters
2. Leukemia in radiologists
3. Thyroid carcinoma in children irradiated in infancy
4. Various neoplastic diseases, including leukemia, in patients receiving therapeutic radiation for a variety of reasons
5. Leukemia in offspring of mothers irradiated during pregnancy
6. Leukemia in survivors of Hiroshima and Nagasaki bombings

C. Life-Span Shortening

1. Studies with various animal species.
2. Radiologists vs. other physicians.

Evaluations of these observations in terms of allowable radiation dose to man is an exceedingly complex task. The National Council on Radiation Protection and Measurements (NCRP) has accepted the task and over the past 30 years has established recommendations regarding maximum permissible doses. Because of the fragmentary evidence currently available, the NCRP points out that these levels are not necessarily completely safe limits but rather levels which should carry an acceptable risk.

In 1959, the Federal Radiation Council was formed to provide a Federal policy on human radiation exposure. This organization published its first report in May, 1960. Its recommendations were essentially the same as those of the NCRP and the International Commission of Radiological Protection (ICRP). The Council suggested, however, that the terms "Radiation Protection Guides" replace the former "Maximum Permissible Doses." The newer terminology recommended by the Federal Radiation Council is used throughout this lecture outline.

III. CURRENT RECOMMENDED DOSE LIMITS

Radiation Protection Guides vary depending on whether the whole body or only a portion thereof is exposed and whether radiation workers or the general public is involved.

A. Radiation Workers

1. ACCUMULATED DOSE (EXTERNAL SOURCES)

a. For external exposure to the whole body, the maximum average dose rate should not exceed 5 rems per, year. The same dose rate applies to the head and trunk, active blood-forming organs, the gonads, and the lenses of the eyes.

No occupational dose is allowed persons under 18 years of age. The accumulated dose at any subsequent age, therefore, should not exceed 5(N-18) rems where N is the age of the worker in years.

b. Generally, however, where only a single portion of the body is exposed, Radiation Protection Guide average dose rates are higher. For example, it is recommended that the average dose rate to the hands and forearms and to the feet and ankles should not exceed 75 rems per year.

2. ACCUMULATED DOSE (INTERNAL SOURCES)

a. Certain elements, upon ingestion, inhalation, etc., deposit uniformly throughout the body. Radioisotopes of these elements deliver a dose to the whole body. The Radiation Protection Guide dose rate for whole-body exposure from such radionuclides has the same limit as from external sources (i.e., 5 rems per yr.). No occupational exposure is allowed persons under 18 years of age.

b. Other radionuclides tend to localize in one or more selected body organs. Based on the essentiality of the organ to proper body function, epidemiological experience with radium, and genetic considerations, other Radiation Protection Guides have been specified for selected critical body organs.

(1) For most individual organs, it is recommended that the maximum average dose rate not exceed 15 rems per year.

(2) For the thyroid and skin, the maximum average dose rate should not exceed 30 rems per year; for the gonads, the average dose rate should not exceed 5 rems per year.

(3) For compact bone, the maximum permissible dose rate is limited to the average dose rate which would result from a body burden of 0.1 microcurie of radium. (This amount of radium would yield a dose rate of approximately 28 rems per year.)

B. General Population Groups

Recommended Radiation Protection Guides for the general population are considerably below those for radiation workers. One of the reasons for this stricter limitation on allowable dose is related to the possibility of genetic effects from radiation exposure. Because the general population contains all of the germ plasm controlling the genetic viability of the race, any mutational effects on this population would have a far greater total impact than exposure of that small segment of the population working with radiation.

Currently it is recommended that the yearly radiation exposure of individuals in general population (exclusive of natural background and the deliberate exposure of patients by dentists and doctors) should be held to one-tenth of the permissible occupational levels. Thus, for whole-body exposure of individuals in the general population, the radiation dose should not exceed 0.5 rem per year.

Exposure of the total population (this includes radiation workers and special groups, as well as the general population) presents a different problem. Here, it is necessary to make assumptions concerning the relationship between the recommended Radiation Protection Guide of 0.5 rem per year for individuals in the general population, and the average dose received by average total population groups. The Federal Radiation Council suggests the use of the arbitrary assumption that the majority of individuals do not vary from the average by a factor greater than three. Thus, they recommend the use of 0.17 rem as a Radiation Protection Guide for yearly whole-body exposure of average total population groups. This is in keeping with their recommendation that the average genetically significant dose to the total population not exceed 5 rems up to age 30 (exclusive of natural background and purposeful exposure of patients by practioners of the healing arts).

Assessment of the amount of radiation dose received by a given individual or population group requires totalling the doses received from external and internal sources. External exposures may be measured with a sufficient degree of accuracy by direct roentgen-rate monitoring procedures. Internal doses are assessed in terms of the concentration of specific radionuclides in air, water, milk, and food which may be inhaled or ingested. To enable comparison of the radioactivity concentrations measured in these various environmental phases with Radiation Protection Guides, concentration values[1] which are related to the established RPG have been derived for many radionuclides.

C. Factors Influencing Radioactivity Concentration Guides

The Radioactivity Concentration Guide is the concentration of radioactivity in the environment which is determined to result in whole-body or organ doses equal to the RPG. In calculating RCG values for a given radionuclide, the following factors must be taken into consideration:

1. INITIAL BODY UPTAKE

Large fractions of some elements are absorbed when taken into the body. In the case of certain other elements, only small fractions are absorbed in passage through the gastrointestinal tract.

Therefore, the greater retention would increase the hazard from the first group as compared with the second, other factors being equal.

When radionuclides are inhaled, unless information specific to the radionuclide is available, it is assumed, in the case of soluble compounds, that 25 percent is retained in the lower respiratory tract. From here, the nuclides move into the blood stream and a portion of each is deposited in its critical tissue within a few days. Approximately 50 percent is held in the upper respiratory tract and is swallowed. In the case of insoluble compounds, it is assumed that 12 percent is retained in the lower respiratory tract, which is usually taken as a critical organ. The remainder is eliminated by exhalation and swallowing.

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[1] Concentration values are expressed in terms of Radioactivity Concentration Guides (RCG) or Maximum Permissible Concentrations (MPC).

2. FRACTION RETAINED IN THE BODY

The rate of elimination from the blood and tissues of the body varies considerably for different elements or compounds. The time required for one-half of the original quantity of radioactive material to be removed from the body by biological processes is called the biological half-life.

Some materials in the blood stream are eliminated rapidly from the body whereas large fractions of others remain in the body organs. For example, radium, plutonium, and strontium are deposited in the bone where the rate of turnover is very slow; i.e., the biological half-life is many years. Radioisotopes of these elements are much more hazardous than those of carbon, sodium, and sulfur which have biological half-lives of a few days or weeks. The principal biological methods of elimination of radionuclides from the body are the urine, feces, exhalation, and perspiration. Usually elimination is much more rapid before the radionuclide is translocated from the body to a more permanent area, such as the bone. This time is usually from a few days to a few weeks. After the initial period, the elimination rate becomes more nearly exponential, and the application of the term "biological half-life" has more meaning.

3. ORGAN OR TISSUE IN WHICH RETAINED[2]

a. Radiosensitivity - Certain body organs or tissues are more radiosensitive than others. For example, the lymphatic tissue and bone-marrow are much more radiosensitive than muscle or nerve tissue. Therefore, in equal concentrations, an element like plutonium is more hazardous than uranium because the plutonium concentrates in the most sensitive part of the bone, whereas the uranium goes to other portions of the bone, to the kidneys, and to other relatively insensitive organs.

b. Size - For a given number of microcuries of a radionuclide in a critical organ, the smaller the organ the greater the concentration and the greater the dose delivered to the critical tissue. Iodine presents a much greater problem than sodium, since the iodine is very selectively absorbed in a much smaller organ, the thyroid gland, whereas sodium is rather uniformly distributed throughout the whole body. In many cases the radionuclide is deposited in a large organ but localized in a small portion of that organ, so that, in effect, the mass of the critical tissue may become very small.

c. Essentiality - Some organs are either not essential to the body function, or, when they are damaged or removed, special steps can be taken to supplement or compensate for their reduced function. For these reasons damage in some cases to the bone-marrow, kidneys, eyes, etc., may represent a greater hazard than equal tissue damage to the thyroid gland.

4. PHYSICAL PROPERTIES OF THE RADIONUCLIDE

a. Type and Energy of Radiation - The hazard is proportional to the relative biological effectiveness or quality factor of the radiation; e.g., alpha radiation is more damaging than beta radiation for a given energy of radiation. As the energy of the radiation is increased, the damage also increases.

b. Half-Life - The shorter the physical or radioactive half-life of a radionuclide, the less is the biological hazard. The radioactive half-life and the biological half-life may be combined into a single "effective" half-life by means of the formula:

$$T_{eff} = \frac{(T_r) \times (T_b)}{T_r + T_b}$$

D. Calculations of Radioactivity Concentration Guides

As an example of the approach used in determining RCG values, the case of cesium-137 is illustrated.

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[2] In certain instances, an organ of the body, such as the GI tract or lung, may receive a significant dose due to the mere presence of the radionuclide in material inhaled or ingested. Under these circumstances, the organ so exposed (even though not absorbing the radionuclide) may become the "criticial" organ.

1. The element cesium is distributed rather uniformly throughout the body. Therefore, an RPG for the radiation worker of five rems per year is selected for any radioisotope of this element.

2. Experimental and theoretical studies have shown that the presence in the body of 30 microcuries of ^{137}Cs will yield a whole-body dose rate of five rems per year. This quantity, 30 microcuries, is referred to as the "maximum permissible body burden" for ^{137}Cs.

3. If, however, 30 microcuries of ^{137}Cs were deposited within the body, they would not remain there long because of biological elimination and radioactive decay. Therefore, a certain amount could be ingested continually without exceeding the maximum permissible body burden.

4. A "standard man" consumes approximately 2200 milliliters of water per day. Approximately one-half of this is ingested during the working hours. If the daily intake of ^{137}Cs via drinking water is to be limited such that the "maximum permissible body burden" will not be exceeded (and the average yearly radiation worker whole-body dose rate not exceed five rems), then the concentration of this nuclide in drinking water must not exceed 4×10^{-4} microcuries per milliliter. This value, 4×10^{-4} Ci/ml, is the Radioactivity Concentration Guide (RCG) for ^{137}Cs in drinking water for consumption by radiation workers, based on an eight-hour per day, 40-hour work week for a radiation worker.

In a similar manner, it can be shown that an average concentration of 6×10^{-8} Ci/ml soluble ^{137}Cs in air in the working environment would result in a worker receiving a whole-body dose rate of five rems per year.

E. Standardization of Calculations

When efforts were first made to establish permissible body burdens and Radioactivity Concentration Guide values for the various radionuclides, it became evident that it would be difficult, if not impossible, to compare suggested values from the different laboratories unless all used the same basic assumptions relative to the average man. Most of the calculations were based on the same fundamental physical assumptions relative to the energy schemes and radioactive half-life values, but each person making these calculations used his own preferred values for the mass and effective radius of the body organs, the rates of ingestion and inhalation, elimination rates, etc. This difficulty was recognized at several major conferences. As a consequence certain characteristics of the so-called "standard man" were agreed upon.

F. Graded System of Actions

It was felt by the Federal Radiation Council that a commitment to the single values represented by the Radioactivity Concentration Guides might prove arbitrary and inflexible, and not provide adequate guidance for taking assessment and control actions appropriate to the risk involved. Therefore, the Council's Report No. 2 introduced the concept of graded scales of RANGES of action in which average daily intake of a radionuclide by samples of exposed population groups are listed in three ranges. These ranges apply to certain selected radionuclides of health significance, namely, radium-226, strontium-89, strontium-90, and iodine-131. In general, the single radioactivity concentration guide equivalent corresponds to the upper limit of RANGE II. For intakes in RANGE I, the Council guideline recommends periodic, confirmatory surveillance as necessary; for RANGE II, quantitative surveillance and routine control is indicated; and for RANGE III, the Council advises evaluation and application of control measures as necessary.

G. Protective Action Guides

All of the foregoing guidelines are predicated upon the more or less constant release of low-level radioactivity into the environment from the routine uses of radiation, and assume continuous radionuclide intake by the population. Control, then, is based primarily at the source. There are cases, however, in which the contamination of the environment might be accidental or unforeseen, producing contamination which is transient and not likely to recur; these might include reactor incidents which result in relatively high but temporary local radioactivity levels, for example. In cases of this kind, the "contaminating event" would not occur on a regular basis, and control, or protective action, might be based upon limiting or changing the intake of certain contaminated foods by the population. However, the impact of such measures upon the community which they are designed to benefit requires careful consideration by responsible authorities to insure that the benefit of the action taken is not outweighed by its other effects. To deal with this kind of situation, the Council, in its Reports No. 5 and No. 7, introduced the Protective Action Guide concept. The PAG is defined as the projected absorbed dose to individuals in the general population which would warrant protective action following a contaminating event. It is assumed that the corresponding projected dose to a suitable sample of the exposed population would be one-third of the PAG. These guidelines have, thus far, been established for iodine-131, strontium-90, and cesium-137. It should be noted that a decision to take action to limit a community's intake of an important basic food item containing radionuclides involves a balancing of the health benefits to be attained against the undesirable features of the protective action, such as disruption of dietary habits and nutritional needs. It follows, then, that control actions could be employed at dose levels above or below the PAG, depending upon the degree of total impact which the action has upon the community.

V. SUMMARY

Based on current information, Radiation Protection Guides for radiation dose have been recommended. These levels are subject to modification as more knowledge is gained. In view of their status and the possibility that any radiation dose may be damaging, it is well to recall the definition of the Radiation Protection Guide as that radiation dose which should not be exceeded without careful consideration of the reasons for doing so. Every effort should be made to encourage the maintenance of radiation doses as far below the Guide as practical.

Table 1 summarizes the current recommendations of the ICRP and the Federal Radiation Council.

Table 1. Radiation Protection Guides

Type of Exposure	Condition	Dose[1] (rem)
Radiation worker:	Accumulated dose	Five times number of years beyond age 18
(a) Whole body, head, and trunk, active blood-forming organs, gonads, or lens of eye	13 weeks	3
(b) Skin of whole body and thyroid	Year	30
	13 weeks	10
(c) Hands and forearms, feet and ankles	Year	75
	13 weeks	25
(d) Bone	Body burden	0.1 µCi of ^{226}Ra or its biological equivalent
(e) Other organs	Year	15
	13 weeks	5
Population:		
(a) Individual	Year	0.5 (whole body)
(b) Average	30 years	5 (gonads)

[1] Minor variations from other recommendations are not considered significant in light of present uncertainties.

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Disposition of Radioactive Wastes

I. INTRODUCTION

Wastes containing radioactive contaminants differ considerably from wastes containing ordinary chemical or biological pollutants. Radioactive contaminants are not perceptible nor objectionable to the senses as are normal polluting substances. And concentrations which might arise from abnormal waste disposition practices would require long periods of exposure before manifestation of any injury.

These wastes are a potential public health problem because continued or repeated exposure by immersion, inhalation, or ingestion may result in accumulation within the body of sufficient quantities of these substances to induce long term harmful effects. The hazard is enhanced by the fact that no treatment except natural decay will reduce the activity of these substances. They cannot be neutralized as are acid wastes or stabilized by biological treatment. All treatment processes, other than natural decay, result in a solid, gaseous, or more concentrated liquid waste containing essentially all the radioactivity in the original waste, the principal objective in such processes being to reduce the volume of wastes that have to be stored or handled.

II. SOURCES OF RADIOACTIVE WASTES

The development of the atomic energy industry has thus far been accomplished under the control or guidance of the U.S. Atomic Energy Commission. As a consequence, the industry has been obligated to minimize its discharge of radioactive materials into the environment so that a health hazard will not result. Radioactive wastes are being controlled by containment and isolation insofar as possible at the present time. The degree of success of this approach has demonstrated that proper control at the potential waste source is a practicable means of limiting environmental contamination problems in populated areas. The treatment and disposal methods used in this control are, in many cases, unique to the industry, and future waste problems may require other novel solutions. The status of waste disposal in the overall industry is summarized below.

A. Nuclear Fuels Production Facilities

Uranium-bearing ores, the raw product of the nuclear industry, are mined principally in the western portion of the United States with approximately one half of the ore being produced in the vicinity of Grants, New Mexico. The radiation problems associated with the mining operations are due mainly to air-borne radioactive dusts and fumes, with comparatively minor liquid wastes resulting. In subsequent milling and refining operations, aerosols and liquid wastes are produced as a result of wash downs and scrubbing processes. The liquid wastes may present a chemical toxicity as well as radioactivity problem and releases of such wastes have led to extensive studies by the Public Health Service in the Animas River area of Colorado and New Mexico (1).

Refined uranium may be processed for enrichment in the U^{235} isotope. Processing plants for this purpose are located at Oak Ridge, Tennessee; Paducah, Kentucky, and Portsmouth, Ohio. Generally, enrichment is accomplished by gaseous diffusion, a process involving physical separation of U^{235} from U^{238} and in which essentially no wastes are produced.

The U^{238} in natural uranium may be converted into Pu^{239} through neutron bombardment in a reactor. The reactors at the Hanford Works in the state of Washington are operated for this purpose. Two types of liquid wastes result from such operations:

1. REACTOR COOLANTS

Dissolved and suspended minerals present in cooling water, circulated through these reactors, are subjected to intense neutron bombardment. As a result, these minerals are made radioactive. Fortunately short-lived radionuclides make up the major portion of the activity so induced. After a short retention period to allow decay, these wastes at Hanford are released to the Columbia River. The health significance of such releases is generally evaluated in terms of drinking water criteria or other stream uses.

2. CHEMICAL PROCESSING WASTES

Plutonium produced in the irradiated uranium is removed by chemical separation. Highly concentrated fission product wastes result from this process. These wastes, the bulk of which are liquid, are stored in permanent holding tanks. Currently, these wastes constitute the most significant waste disposition problem arising from the nuclear reactor industry.

B. Basic Research Programs

The Atomic Energy Commission sponsors a variety of basic research programs in chemistry, physics, biology, and other allied fields at numerous sites throughout the country. Resulting from such studies, a myriad of types of radioactive waste materials are produced including laboratory drainage and wash waters, laundry wastes, contaminated equipment, etc.

Gaseous or air-borne wastes are usually passed through filters to remove radioactive particulate matter. Radioactive gases are normally discharged into the atmosphere in a manner such that maximum permissible concentrations for breathing are not exceeded.

Solid wastes are usually disposed of in controlled burial areas or in the ocean. Combustible wastes may be concentrated by incineration prior to burial.

Liquid wastes comprise the principal disposal problem. Low-level wastes (radioactivity content in the order of $10^{-4}\mu c/ml$ or lower) are usually disposed of by dilution. For higher-level wastes, or where sufficient dilution capacity is not available, the radioactive constituents are separated from liquid, ordinarily by evaporation or by passage through ion exchange material. The concentrated radioactive material is then stored with the decontaminated liquid being released to the environment.

C. Experimental Reactor Program

Many research reactors have been constructed in recent years. Certain of these reactors are designed for further study of economical nuclear power production. Research reactors designed essentially to provide a source of neutrons for various experimental studies are also in operation.

Reactors for research purposes per se are the origin of a comparatively small pollution problem. Under current practice, when the fuel mixture becomes expended, processing of this mixture and disposal of the waste fission products is handled at central processing sites such as at Idaho Falls, Idaho, and Hanford, Washington.

D. Nuclear Power Reactors

The heat produced in a reactor determines the power or electrical energy which may be generated. Since heat production is proportional to the amount of fuel burnup and consequently the amount of fission products formed, the quantity of radioactive material produced can be related to the power output for a given reactor. The fission product concentration at equilibrium in a reactor has been estimated to be about 6 curies per watt of heat produced. Assuming an efficiency of 20% in converting this heat into electrical energy and assuming used fuel is permitted to decay for 100 days prior to processing, the fission product activity resulting from one year's operation of a nuclear reactor would amount to about 0.7 curie per watt of electrical energy generated.

The rate of demand for electricity in a city of 50,000 population is approximately 20 megawatts. If this power were furnished by a 100 megawatt (heat output) nuclear reactor, operation for one year would yield some 14 megacuries of fission products at 100 day's decay. Assuming a fission product concentration of 10^4 to 10^6 $\mu c/ml$ in the wastes resulting from the chemical processing of the fuel from this reactor, there would result an annual production of 4,000 to 400,000 gallons. A waste of such high concentration would, of course, require leak-proof storage. Its potency can be appreciated if it is considered that 1 drop of a waste containing 10^6 $\mu c/ml$ would require dilution in approximately 1 billion gallons of water to lower the concentration to 10^{-8} $\mu c/ml$.

The radioactivity in such a waste will decrease with time through decay of the various radionuclides to stable products. For the hypothetical reactor, the 14 megacuries of activity remaining at 100 days after removal from the reactor would decay to approximately 3.7 megacuries at 1 year and to 0.6 megacuries at 5 years. At this time (5 years) the contribution of $Sr^{90} + Y^{90}$ and $Cs^{137} + Ba^{137m}$ would represent approximately two-thirds of the total activity. Inasmuch as Sr^{90} and Cs^{137} have half-lives of about 30 years, storage of such waste must be for considerable periods. It has been estimated that storage for 6 centuries will be required prior to release of power reactor wastes into the environment.

E. Radionuclide Distribution Program

The distribution of radionuclides to private organizations such as hospitals, universities, and industrial firms is becoming more widespread. Since the inception of the program in August, 1946, more than 4,500 organizations in the United States (together with a number of groups in foreign countries) have received shipments of these materials. To date, over 120,000 shipments, containing in excess of 600,000 curies, have been made.

In December 1956, the amount of radioactivity distributed in domestic shipments, excluding shipments to AEC facilities, totalled 198,464 curies. Approximately 90% of this total was comprised of cobalt-60 and iridium-192, which are used in encapsulated form and are therefore never likely to be discharged in liquid waste. Of the remaining, which constituted some 20,000 curies, about 25% was iodine-131 and phosphorus-32.

On a curie basis, the overall magnitude of the waste disposal problem associated with radionuclide users is very small in comparison with that associated with the atomic energy industry itself. Nonetheless, wastes from users pose unique problems in many instances. These problems are magnified in that users may not be experienced in waste disposal techniques, and special radioactive waste treatment facilities are not usually available. In addition, a number of individual users in a community may be using the same sewerage discharge for dilution purposes, in which case the additive effects of such wastes may result in a potential health hazard. Almost half of such users, for example, discharge their liquid wastes to the sewer.

III. WASTE DISPOSAL CRITERIA

With regard to the peacetime pursuits of the atomic energy industry, the release of radioactive waste materials into the environment is subject to man's control. From the public health standpoint, each of the various types of radioactive wastes discussed above may be regarded as a potential environmental contaminant. As radionuclide users in medicine, industry, and research increase, the probabilities of inadvertent discharge to the environs will also increase. For high-level wastes presently being stored in tanks, tank deterioration with age, damage due to earthquake, or damage due to sabotage or bombs during wartime, represent possible eventualities which must be considered.

Generalized waste disposal criteria have been developed by the National Committee on Radiation Protection and Measurements (2). The principal criterion states: "Users of radioactive material shall release these materials only in such a manner that the radioactive material discharged, in combination with that discharged by other users, will not cause contamination of the environment that may result in a person or persons receiving an excessive radiation dose". The average concentration of a radionuclide in air or water in the neighborhood of atomic energy installations should not exceed 10% of the maximum permissible levels recommended for radiation workers. (3).

Because the above criteria apply to point-of-use, users may take advantage of such factors as decay, dilution and dispersion in the environment to enable higher levels of activity to be released at the point-of-discharge. In this regard, specific waste disposal criteria have been developed by the NCRP for discharge of iodine-131 and phosphorus-32 into public sewers (4) and for discharge of carbon-14 into the atmosphere and into sewers (5). Discharge criteria for other radionuclides have been developed by the Division of Licensing and Regulation of the Atomic Energy Commission and have been published as Title 10, Part 20, in the Federal Register. (6)

Other specific criteria have been developed by the National Committee on Radiation Protection and Measurements relative to ocean disposal of radioactive materials. (7) A forthcoming report of the NCRP, "Burial of Radioactive Wastes in Soil", is expected to specify criteria for disposal by burial.

IV. METHODS OF DISPOSITION

At present, there are two general philosophies governing the disposition of radioactive wastes. The first is the philosophy of dilution and dispersion. That is, the reduction of the concentration of radionuclides by carrier dilution or dilution in receiving medium to the point where the nuclide or mixture can no longer be reconcentrated to hazardous levels. Disposition by this method depends upon the characteristics of the wastes and the receiving body. Where dilution and dispersion are not feasible, the second philosophy of radioactive waste disposition is utilized; this is the process of concentration and confinement. Such procedures entail the reduction in volume of wastes and subsequent storage for decay purposes. The treatment methods discussed are actually methods of concentration which may be grouped under this second philosophy.

A. Air-Borne Wastes

The two philosophies of waste disposition for radioactive materials can be applied to the treatment of air-borne wastes through ventilation (dilution) and decontamination (concentration). Ventilation would include discharge from hooded operations to the atmosphere under conditions which provide dilution to safe limits. In venting wastes through stacks, the height of the discharge stack is established on the basis of meteorological conditions in the area, the distance to areas to be protected, and whether or not the plant must be operated continuously or intermittently.

Decontamination processes vary, depending principally on the physical state (dissolved or particulate) of the contaminant. Radioactive gases produced as fission products include iodine, krypton, and xenon. Iodine can be removed by scrubbing or by reaction with silver nitrate. The noble gases may be removed by complex adsorption processes. With air-cooled reactors, the significant components of air which become radioactive are argon (forming argon-41) and to a minor, nitrogen (which yields carbon-14). Normally, these can be handled by dilution techniques.

Particulates in air-borne effluents may contain significant amounts of radioactivity. Standard separation techniques are used for removal of these particulates. The techniques include the use of settling chambers, filters, scrubbers, mechanical or centrifugal separators, and electrostatic and thermal precipitators. Where the temperature of exhaust gases is lower than 275°F, high efficiency filter media of craft paper and asbestors fiber in wooden frames are used. If air temperatures are higher, the filter media used are made either of all glass fiber or of glass and asbestos fiber, housed in metal holders. These high efficiency filters have a removal efficiency of 99.95% (decontamination factor of 2,000) for particulate material as small as 0.3 microns.

The concentrated radioactive materials resulting from these decontamination processes are commonly in solid form and disposed of usually by burial.

B. Liquid Wastes

Methods for the disposition of liquid wastes includes dilution for low-activity wastes and storage for high-activity wastes. Intermediate level wastes are treated by various physical, chemical or biological methods which serve to separate the waste into a relatively non-radioactive portion which can be disposed of by dilution and a high activity portion which must be stored. These treatment processes merely serve to reduce the volume of material for storage.

Dilution in sewers and streams may be accomplished either with the volume of water as a dilutent or by using stable counterparts of the radioactive isotopes as diluting material. The criterion governing this procedure is to reduce the concentration of radioactive material to that point where it can no longer constitute a significant hazard either directly to humans or indirectly through aquatic life. This dilution must take into consideration the concentration of certain materials that takes place in various forms of aquatic life. Algae, for instance, have been known to concentrate radiophosphorus by factors as high as 10^5.

At the present time, it is customary to store most high-level radioactive waste residues in underground tanks pending development of more economical storage or recovery methods. The cost of tank storage varies from 35¢ to $2.00 per gallon for the tanks alone. Hence, the use of treatment processes to reduce the volume of wastes to be stored is, in certain cases, a necessary adjunct to the handling procedure. The evaporators designed for this purpose must include devices to prevent entrainment and carryover of radioactive contaminants to the distillate. Evaporation costs have been estimated at between 6 and 32¢ per gallon of waste handled. Therefore, evaporation cannot be economically used for high-volume low-activity wastes.

Small volumes of waste liquids may be mixed with cement to form a concrete which can be buried underground or at sea. Ground or sea burial must be accomplished with consideration of various environmental factors. In ground burial, conditions of ground water, soil acidity or alkalinity, etc., must be noted as well as long-term control being maintained over the burial site.

Sea burial has been utilized on both the Atlantic and Pacific coasts. "In the Atlantic, the major disposal areas are 230 miles and 150 miles southeast of Sandy Hook, both off the continental shelf at a depth in excess of 1000 fathoms. About 8000 curies plus the reactor structure of the dismantled Sea wolf prototype, which is estimated at 33,000 curies induced in stainless steel, have been dumped to date." (8)

"In the Pacific, about 14,000 curies has been dumped at a site 48 miles west of the Golden Gate at a depth of over 1000 fathoms." (8) At the present, however, many experts believe that too little is known about oceanographic conditions to permit wide-scale dumping of radioactive wastes at sea. Since, when this is done, long-term control over the wastes is relinquished.

Chemical methods in current use for the concentration of radioactive materials in liquid wastes include: carrier precipitation or coagulation, ion exchange, and adsorption on selected media.

Relatively high concentrations of radioactivity in liquids may represent extremely low weight concentrations, i.e., low "parts per million" content. As a consequence, precipitation of carrier-free radionuclides is usually impracticable. A technique utilized is to precipitate a stable isotope of the particular radionuclide and thus carry down the radionuclide as well. This process, known as carrier precipitation, can also be used for the separation of various radio elements such as strontium, etc. A similar process in which a preformed precipitate is utilized to carry down soluble radionuclides is the process of coagulation which is familiar to water treatment personnel. In this latter process, alum or iron salts are employed. The extent of removal depends upon the chemical form of the radioactive material and other constituents of the waste. Those nuclides which are complexed by complexing agents or detergents may not be effectively precipitated by these processes. In any case, the radioactivity is transferred from the liquid phase to a sludge which must be further processed, and represents a disposal problem in itself.

Ion exchange is utilized for the treatment of large-volume, low-activity wastes where the radiomaterial is present in ionic form. Ion exchange resins, widely used for water demineralization, are very effective in the decontamination of waste liquids, although the regeneration of these resins may produce a liquid waste of considerable volume. For this reason, exchange beds used for the decontamination of radioactive waste liquids are usually run to exhaustion and then treated as a solid waste. Decontamination factors of as high as 10^4 have been achieved using ion exchange resins, although decontamination of fission product mixtures may be considerably less due to the presence of colloidal materials. Adsorption on various surface-active agents has been employed for the decontamination of large-volume low-level waste liquids. These agents include activated carbon, metallic powders such as aluminum and iron, and clays of particular types. The use of clays for the decontamination of waste liquids has been widely investigated since it may offer a cheap medium for the concentration and confinement of materials removed from waste solution. It has been found for example, that certain clays such as montmorillonite and kaolinite, have the ability to adsorb relatively large quantities of radioactive materials from solution and upon sintering can be so changed in crystalline structure as to retain indefinitely the adsorbed material. The adsorptive properties of clay have been utilized for several years in seepage pits. Moderate-activity wastes are emptied into these pits, allowing the liquid phase to percolate through the soil, while retaining in the immediate vicinity of the pit the radioactive constituents adsorbed onto the clay particles.

The utilization of biological treatment techniques has been found to be feasible in several cases, particularly where the radioactive material is bound with a complexing agent or detergents (as in laundry wastes). A trickling filter for example, is in use at the National Reactor Testing Station to concentrate radioactive materials from laundry wastes resulting from the decontamination of clothing worn by radiation workers. The activated sludge process is also being used to treat liquids in which the material, in this case a heavy metal waste, is in a complexed form. These complexes would not be removed by chemical treatment but can be broken down by biological action and the radioactive material recovered with the biological sludge. Decontamination factors for biological treatment are generally low, usually not exceeding 100.

C. Solid Wastes

Solid wastes may consist of laboratory glassware and other equipment used in processing radioactive materials, clothing worn by radiation workers, lumber and other materials from demolished buildings in which radioactive materials were handled, animal carcasses used in biological studies, etc. Where practical, these materials are decontaminated and reused. In most instances, a liquid waste is produced in such decontamination processes. Combustible materials which cannot be decontaminated are often burned. Care must be exercised however, to prevent the escape of excessive volatile and particulate activities.

Where the material cannot be decontaminated or burned, storage for decay or storage of permanent nature (for long-lived radionuclide contaminants) is utilized. At present, the Atomic Energy Commission operates several burial and storage areas for this purpose. One such site is at the Oak Ridge National Laboratory in Oak Ridge, Tennessee. Burial of dense materials has also been accomplished at sea. Generally such materials are encased in concrete prior to dumping.

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*Available from the Superintendent of Documents, U.S. Government Printing Office, Washington, D.C. 20402

GPO 873-936

American Standard

Radiation Symbol

Sponsor
Atomic Industrial Forum, Inc.

Approved March 1, 1960
AMERICAN STANDARDS ASSOCIATION
INCORPORATED

Training Publication No. 239n

PREFACE

Early in 1957, the new Nuclear Standards Board of the American Standards Association authorized a project designated as N-2 for the development of general and administrative standards in the field of nuclear energy. The Atomic Industrial Forum, Inc. agreed to be the sponsor for this project. The scope of the project reads as follows:

> "Standards, specifications and methods of administration associated with the peaceful use of nuclear energy, including color codes, symbols, nomenclature, qualifications of professionals, records and reporting systems and procedures, and accountability of materials."

The organizations which participated in this work and the names of their representatives, as listed at the time the standard was submitted to the sectional committee for approval, are as follows:

Chairman W. A. McAdams, Atomic Industrial Forum
Secretary S.J. Harris, Atomic Industrial Forum

Organization Represented	*Name of Representative*
American Chemical Society	W. W. MEINKE
American Conference of Governmental Industrial Hygienists	M. R. ZAVON
American Industrial Hygiene Association	J. A. MARTIN
	J. W. VANDERBEEK *(Alt.)*
American Municipal Association	E. C. MESTER
American Nuclear Society	S. TOWNSEND
	H. S. MYERS *(Alt.)*
American Society of Mechanical Engineers	H. F. WALCHLI
Association of Casualty and Surety Companies	M. A. SNELL
Association of State and Territorial Health Officers	D. BERGSMA
Health Physics Society	L. ROGERS
International Association of Governmental Labor Officials	M. KLEINFELD
	A. ABRAHAMS *(Alt.)*
National Association of Mutual Casualty Companies	H. T. WALWORTH
National Bureau of Standards	O. H. HILL
National Safety Council	A. J. STEVENS
U. S. Atomic Energy Commission	F. R. ZINTZ
U. S. Department of Defense	C. L. MILLER
U. S. Public Health Service	D. W. MOELLER
	D. R. CHADWICK *(Alt.)*

At its first meeting on July 10, 1957, the Sectional Committee discussed the need for an American Standard Symbol to denote the presence of ionizing radiation, and reviewed a preliminary draft of such a standard. Discussion of the standard continued at subsequent meetings and in late 1958 a subcommittee was appointed to develop a final draft for letter ballot. The personnel of this subcommittee are as follows:

Chairman - F.R. ZINTZ	U.S. Atomic Energy Commission
O.H. HILL	National Bureau of Standards
L.R. ROGERS	Health Physics Society
H.F. WALCHLI	American Society of Mechanical Engineers

AMERICAN STANDARD RADIATION SYMBOL

1. SCOPE

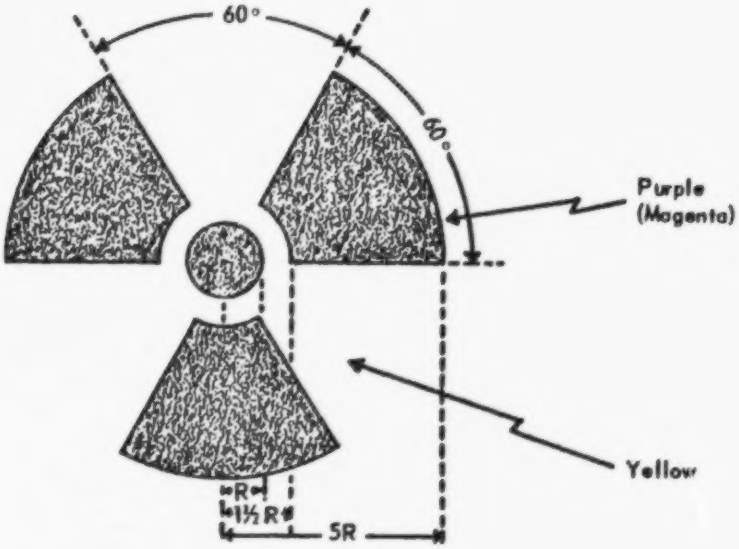
The radiation symbol specified in this standard shall be used to signify the actual or potential presence of ionizing radiation and to identify objects, devices, materials or combinations of materials which emit ionizing radiation. For the purpose of this standard, ionizing radiation includes gamma and X-rays, alpha and beta particles, high speed electrons, neutrons, protons, and other nuclear particles; but not sound nor radio waves, nor visible, infrared, or ultra violet light. This standard does not specify the radiation levels at which the symbol is to be used.

2. STANDARD RADIATION SYMBOL

2.1 The Standard symbol for signifying radiation or radioactive materials shall be designed and proportioned as illustrated in Figure 1.

Figure 1

R = Radius of central disc



2.2 The symbol shall be as prominent as is practical, and of a size consistent with the size of the equipment or material to which it is affirmed or attached, providing that the proportions shown in Figure 1 are maintained, and in any case that the symbol can be read from a safe distance.

2.3 Except when circumstances do not permit, the symbol shall be oriented with one blade pointed downward and centered on the vertical axis.

3. COLOR CODE FOR SYMBOL

3.1 The four parts of the symbol, i.e., the three blades and the center disc shall be in a reddish purple (magenta) color similar to that which American Standard Safety Color Code, Z 53.1, has established for designating the presence of radiation and radioactive materials.

3.2 The symbol shall be located in a yellow background, the yellow having a color similar to that which American Standard Safety Color Code, Z 53.1, has established for use with the purple color to designate the presence of radiation and radioactive materials.

4. RESTRICTIONS ON THE USE OF SYMBOL

4.1 The radiation symbol shall be used or displayed only to signify the actual or potential presence of ionizing radiations as provided in the scope of this standard.

4.2 Appropriate wording used in association with the radiation symbol to indicate the nature of the source of radiation, type of radiation, limits of occupancy and similar precautionary information, shall not be superimposed on the symbol.

Universal Decimal Classification 62-777.6:614.8:539.16

GPO 875-764

Properties and Production of X Rays

I. INTRODUCTION

X rays were discovered in 1895 by Wilhelm Conrad Roentgen, during the course of his experimentation with a gas-discharge tube enclosed in a black paper box. When an electric current was passed through the tube, a faint light was observed to come from a nearby cardboard screen which had been coated with barium platinocyanide. The light from the screen was completely unexpected. After further investigation, Roentgen concluded that this phenomenon was caused by the presence of invisible rays, capable of penetrating many opaque materials and producing fluorescence in certain substances. Roentgen called these penetrating radiations x rays.

II. NATURE OF X RAYS

X radiation is a part of the electromagnetic spectrum, which also includes visible light, radio waves, gamma rays, and infrared and ultraviolet radiations. The distinction among the various radiations is fundamentally one of energy. All electromagnetic radiation propagates through vacuum at 3.0×10^8 meters per second. This speed is the value of the fundamental constant c, commonly called the speed of light.

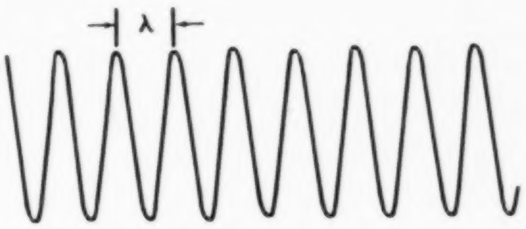
Experimental evidence shows that electromagnetic radiations display two seemingly contradictory properties:

1. Electromagnetic radiations (especially those of low frequency) can often be treated as though they propagate through space as waves.
2. Paradoxically, high-frequency radiations often exhibit properties attributable to transmission as particles. These particles are called photons.

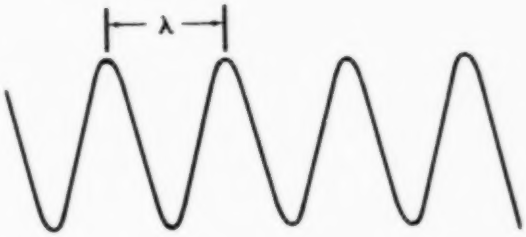
Each of these aspects is important to the complete understanding of the behavior of all electromagnetic radiations, including x rays.

Frequently, to help visualize wave behavior, x rays are said to move like waves on a body of water. While such is not the case, certain characteristics are common to both wave motions. A description of the properties of electromagnetic waves include three related parameters: wavelength (symbol: λ), frequency (symbol: ν), and speed of propagation in vacuum (symbol: c).

Wavelength is the distance between two similar points on two successive waves, as illustrated in Figure 1.



Short Wavelength–High Frequency



Long Wavelength–Low Frequency

Figure 1.–Wavelength–Frequency

Frequency is a measure of the number of waves which pass a given point per unit time. The special unit of frequency is the hertz (Hz). If 1 complete wave in a wave train passes a point in 1 second, the wave train has a frequency of 1 Hz. In other words, 1 Hz equals 1 cycle per second.

It can be seen that if the wavelength becomes shorter, more waves would pass the observer in a given time; i.e., the frequency would go up. Thus, frequency is inversely proportional to wavelength. The three parameters are related by:

$$c = \nu\lambda \qquad (1)$$

or

$$\lambda = c/\nu \qquad (2)$$

here c is the speed of light in meters per second, ν is the frequency in hertz, and λ is the wavelength in meters.

The quantum of electromagnetic radiation (photon) has a frequency and wavelength associated with it and related to its energy by the equation:

$$E = h\nu \qquad (3)$$

or

$$E = hc/\lambda \qquad (4)$$

where E is the photon's energy in joules, h is Planck's constant (6.62×10^{-34} joule seconds), and the other parameters are as previously defined. The relation shows that, as the energy of the photon increases, the frequency associated with the photon increases. Since the wavelength is inversely proportional to the frequency, the wavelength is also inversely proportional to the energy.

It is sometimes difficult to visualize electromagnetic radiations as possessing both wavelike properties and particulate properties. Fortunately, both properties are not observable in the same experiment. Thus, the nature of the experiment can dictate the proper approach. The deeper philosophical significance of the so-called duality of electromagnetic radiation need not be considered here.

The entire electromagnetic spectrum can be divided into various sections classified on the basis of frequency, wavelength, or energy, as shown in Figure 2. The divisions are arbitrary; in fact, most areas overlap with adjacent regions. For example, x rays and gamma rays are for historical reasons distinguished, on the basis of their origin, and not particularly on the basis of their energy.

Classically, electromagnetic radiation is considered a combination of oscillating electric and magnetic fields moving through space. The quantum electrodynamic view inclines more toward considering electromagnetic radiation as streams of photons moving through space. These apparently opposing views constitute the duality previously mentioned. From either point of view, no mass or electrical charge is associated with the radiation; i.e., photons possess neither rest mass nor charge. These radiations, therefore, are not deflected by magnetic or electrostatic fields. In addition, x rays unlike visible light cannot be focused with lenses in a practical sense. In short, x rays, once emitted, will travel in straight lines until they interact with matter.

Electromagnetic radiation in the infrared and higher range of energies can produce latent images on photographic films. This property makes possible various radiographic procedures.

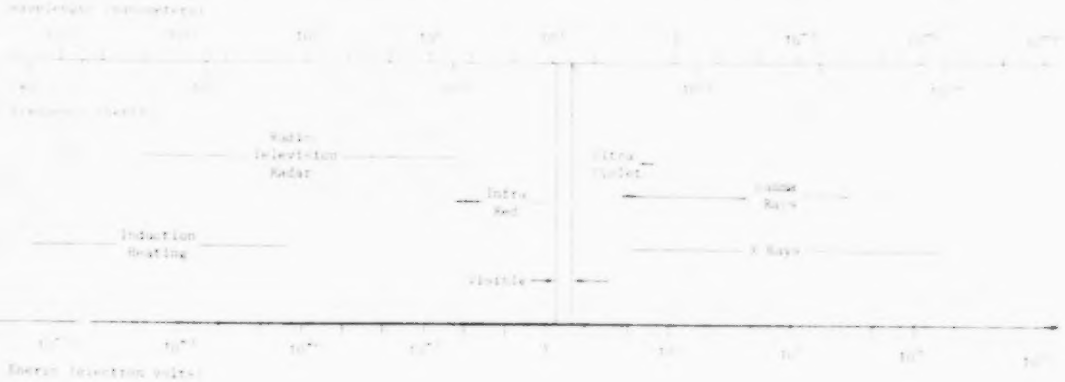


Figure 2. Chart of the Electromagnetic Spectrum

In summary, the properties of x rays generally of interest are:

1. X rays can be considered electromagnetic radiations of short wavelength, high frequency, and high energy.
2. X rays travel in straight lines until they interact with matter.
3. X rays propagate through space at the speed of light.
4. X-ray photons possess no rest mass and no charge.
5. X rays are capable of exposing photographic films.

III. PRODUCTION OF X RAYS

X rays are produced by bombarding a material (the target) with energetic electrons. These electrons lose energy by one or more of three processes:

1. Interaction with the orbital electrons of target atoms, causing their ionization.
2. Interaction with the orbital electrons of target atoms, causing their excitation.
3. Interaction with the coulomb field of target nuclei, resulting in bremsstrahlung (German for "braking radiation").

Only a small fraction of the electrons in the beam are effective in producing x radiation. It has been shown experimentally that only on the order of one percent of the electrons produce radiation in the diagnostic x-ray energy region of the electromagnetic spectrum. Most electrons dissipate their energy in low-energy interactions, ultimately resulting in the production of heat in the target.

The third process (to be discussed first) gives rise to photons of a continuous range of energies. As mentioned, the radiation resulting from this process is called bremsstrahlung. The other two processes produce photons (called characteristic x rays) of discrete energies, giving rise to a sharp line spectrum superimposed on the continuous spectrum.

A. The Continuous Spectrum

Generally, electromagnetic radiation is produced whenever a charged particle is caused to change its speed or direction of motion. The magnitude of this acceleration (here defined as a change in speed, either faster or slower, and/or a change in direction of motion) determines the energy of the radiation produced. This process is responsible for radio-wave emission from antennae, synchrotron radiation, electromagnetic emission from pulsars, and the continuous x-ray spectrum. These radiations are all forms of bremsstrahlung. Electrons are the lightest charged particles, and thus are relatively easily swerved from their courses, or slowed down. For this reason electron acceleration is usually responsible for bremsstrahlung.

When the magnitude of the electron acceleration is great enough to produce photons in the energy range of about 1 keV or higher, the radiation is called x radiation. It is fairly obvious that electrons must have at least 1 keV of energy in order to undergo a large enough acceleration to produce 1-keV photons.

When the bombarding electrons reach the target, some electrons are accelerated by the coulomb fields of the nuclei of the target material. An electron can lose any portion of its energy in one of these nuclear encounters. However, it usually requires several interactions (by any or all of the processes previously listed) for an electron to lose all its energy. The end result is a stream of photons ranging in energy from zero up to the maximum electron energy. Figure 3 shows the resulting idealized bremsstrahlung spectrum.*(1)* As might be expected from the nature of their formation, the low-energy photons are more numerous than the high-energy photons.

The x rays of very low energy are removed from the useful beam, because they do not possess sufficient energy to escape from the target or to penetrate the walls of the x-ray tube. The resulting practical bremsstrahlung spectrum is also illustrated in Figure 3.

To utilize these phenomena, a practical x-ray generator requires:

1. A source of electrons--usually a heated tungsten filament.
2. A source of high voltage to impart high energies to the electrons.
3. A target for the electrons to strike, so their velocity can be changed abruptly.

The target should have a high atomic number (a high electric charge on the nucleus) for more efficient bremsstrahlung production. Target material should

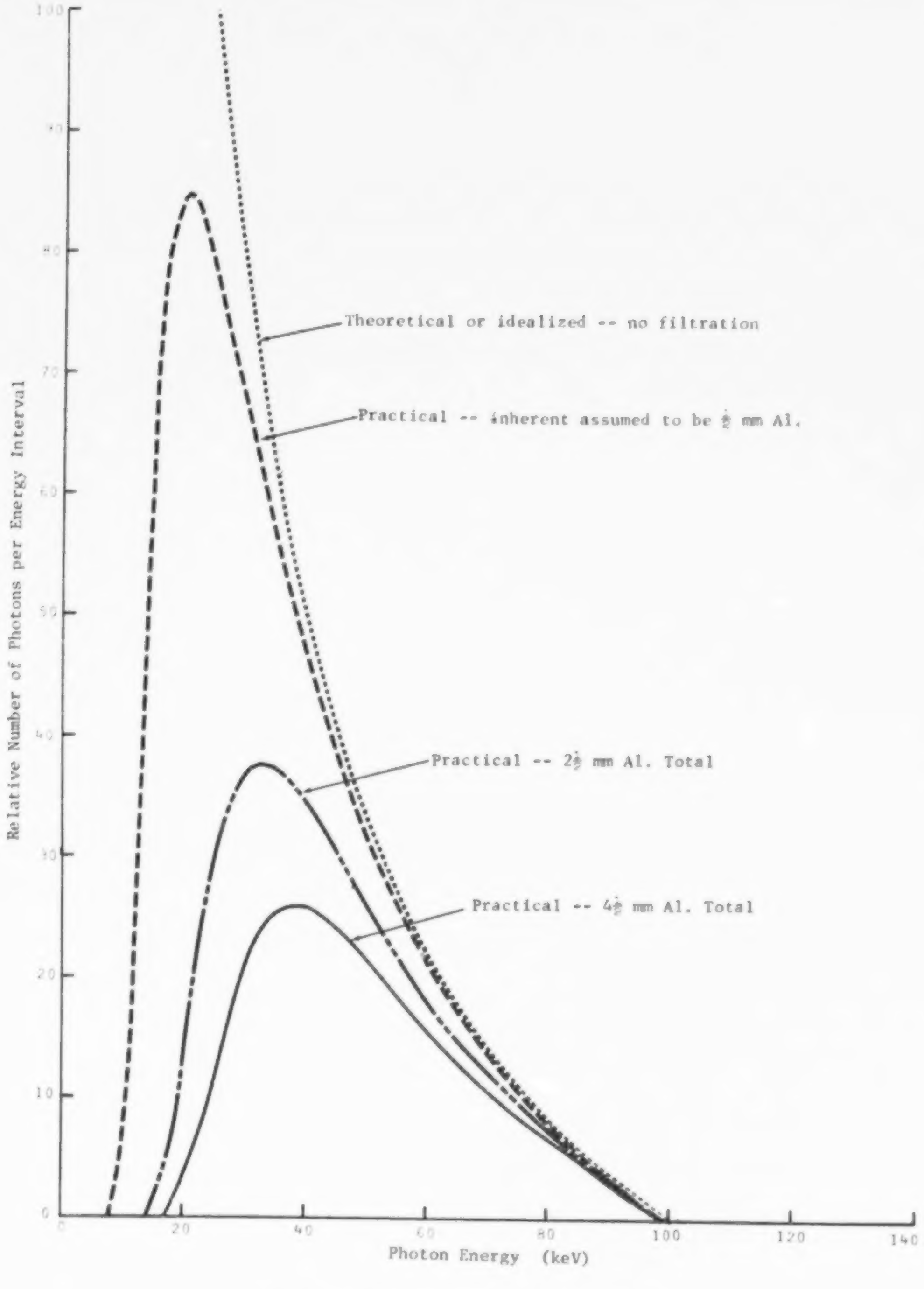


Figure 3.–Bremsstrahlung Spectra–100 kVcp

also have a high melting point, because so much of the electron beam's power is dissipated as heat. In fact, target heating is a severe problem; it is the principal limitation on the output obtainable from x-ray tubes. The use of materials with high melting points, rotating targets to distribute the heating over larger areas, and water- or oil-cooling of the targets help alleviate this problem to some extent. Tungsten, which has a high atomic number and a high melting point, is commonly used as a target material. A stationary target x-ray tube is shown in Figure 4.

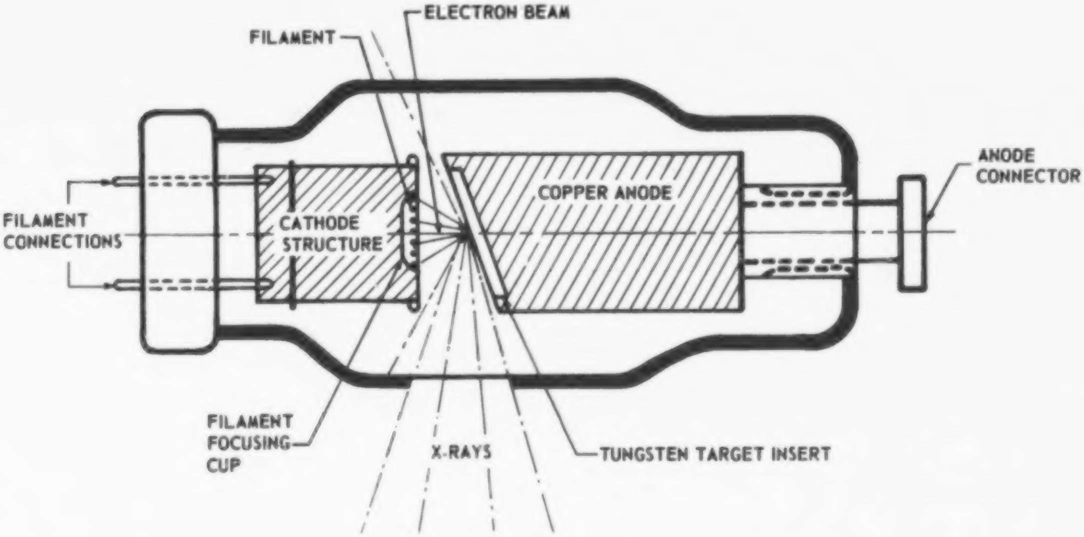


Figure 4.–Stationary Target X-Ray Tube

B. The Line Spectrum

Important additional contributions to the x-ray spectrum result from high-level excitation and ionization of the target atoms. The most tightly bound electrons in an atom are those in the inner [illegible] K) shell; these K electrons are hardest to rem[illegible] the electron beam in the x-ray tube possesse[illegible] a certain energy (about 70 keV for a tungsten target), it can cause K electrons in the target atoms to move to higher shells (excitation) or to move completely out of the atoms (ionization).

Less energetic electrons can perform similar operations on the electrons of less tightly bound shells (L, M, N, etc.). After the atom is excited or ionized, a vacancy exists in one of the atomic shells. The vacancy is immediately filled by an electron from a higher energy state, or by a free electron. Since the energy levels of an atom are discrete and well-defined, electronic transitions from higher to lower levels give rise to photons of discrete energy. These photons constitute the line spectrum, which is superimposed on the continuous bremsstrahlung spectrum. Since each element has its own distinguishing set of energy levels, the x-ray line spectrum produced is characteristic of the target material. For this reason the line spectrum is sometimes called the characteristic x-ray spectrum. The most prominent tungsten K lines are shown in Figures 5 and 6.

These "lines" are not shown to scale. The number of photons in the line spectrum from an x-ray machine with a tungsten target operating at 100 keV (constant potential) constitutes about ten percent of the total photon spectrum.*(2)* In other words, the area under the line spectrum should be shown as about ten percent of the area of the entire spectrum (continuous plus characteristic). Because of their extreme narrowness, the lines would have to extend far beyond the figure in order to show this part of the spectrum in proper scale

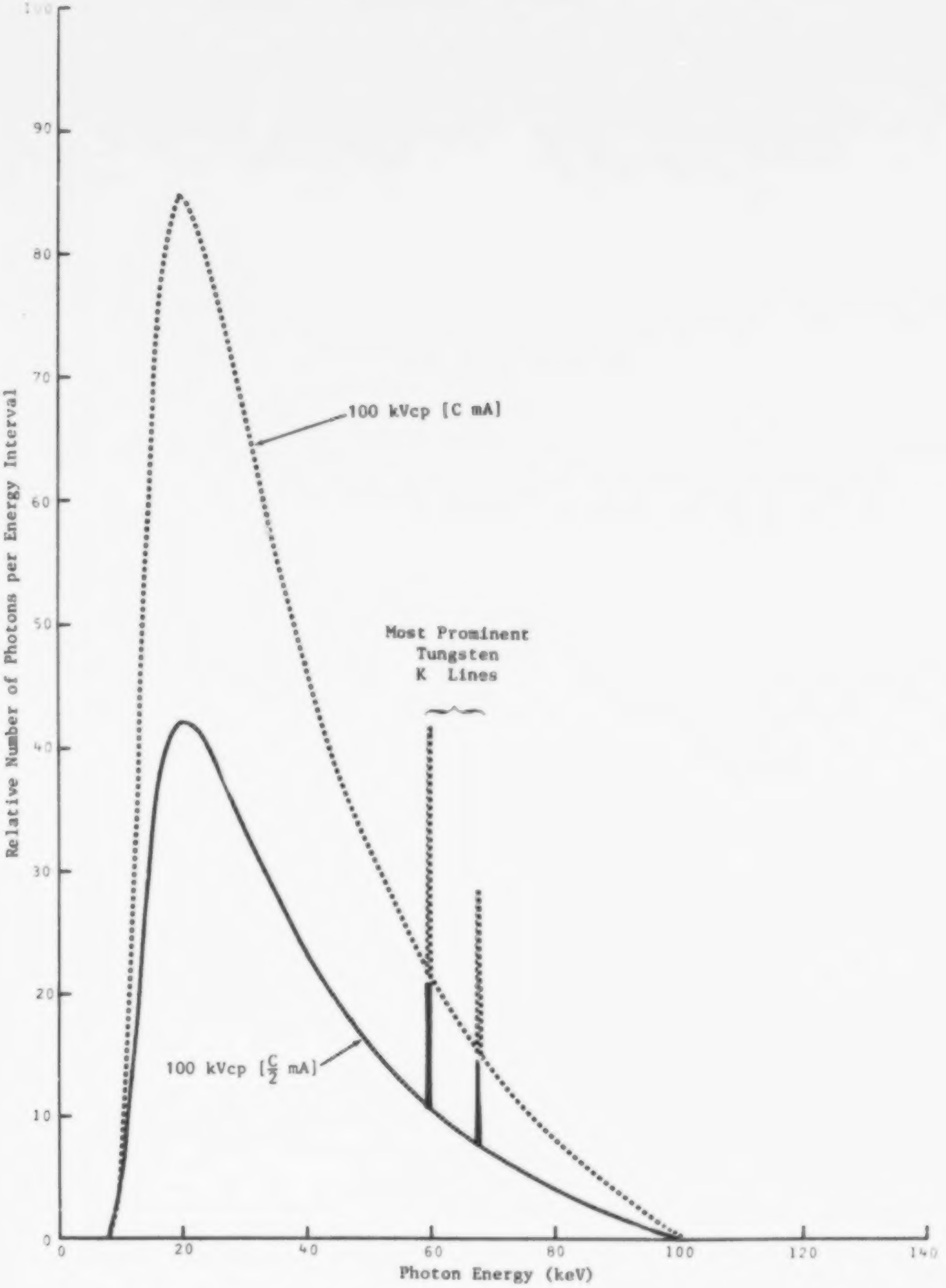


Figure 5.–Effect of mA Change

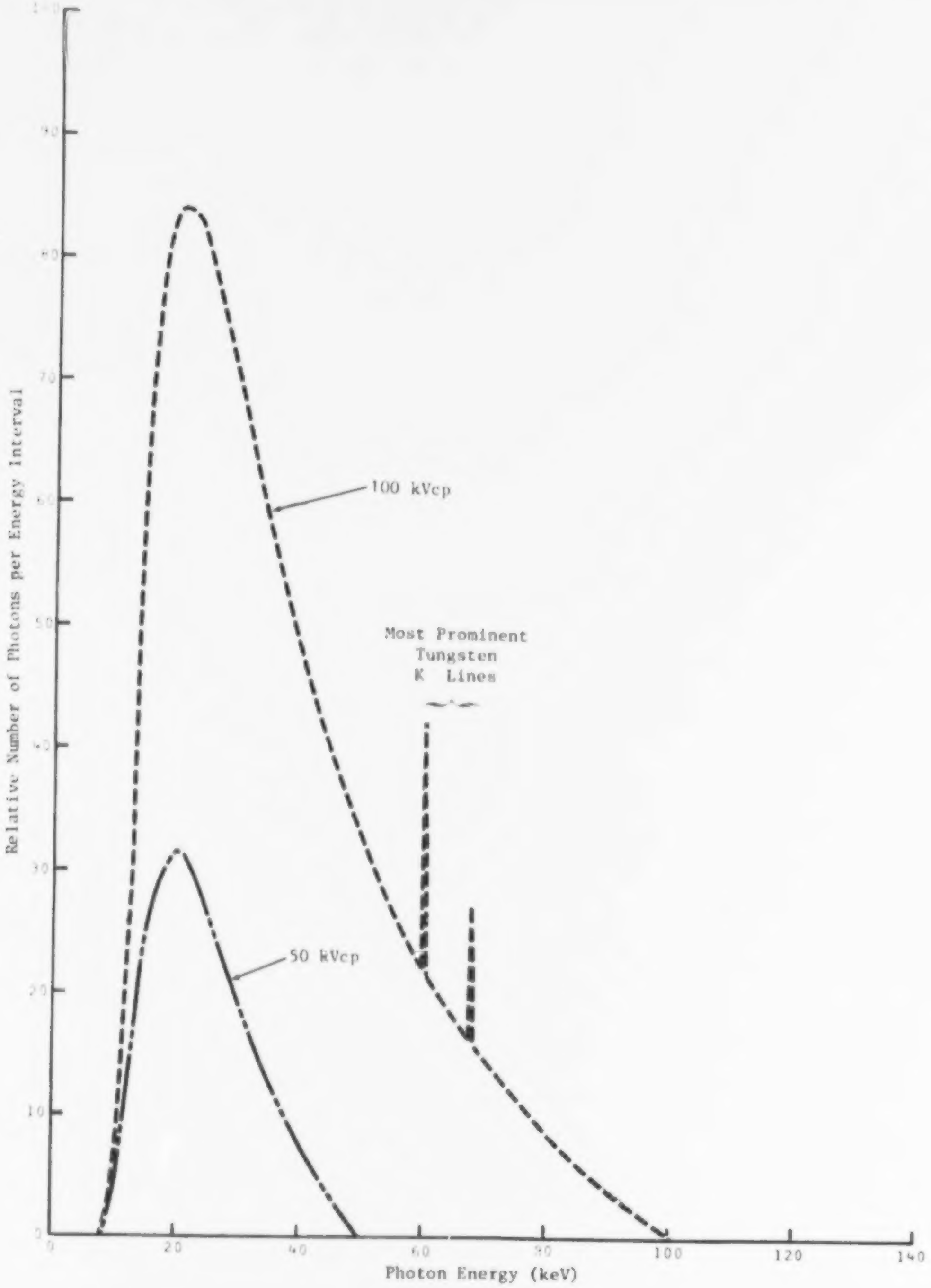


Figure 6.–Effect of kV Change

IV. DISTINCTION BETWEEN X RAYS AND GAMMA RAYS

X rays and gamma rays comprise overlapping portions of the electromagnetic spectrum; their distinction is not necessarily one of energy. In fact, a gamma ray of a given energy is identical to an x ray of the same energy. The distinction between them is one of origin: a gamma ray is emitted from the nucleus of an atom when the nucleus undergoes a transition from a higher to a lower energy level. Gamma rays are thus nuclear analogs of the characteristic x ray produced by atomic de-excitations. Therefore, gamma rays are always of discrete energies determined by the difference in energy between the nuclear energy states which produced them. In contrast, x rays have a continuous energy distribution in addition to their line spectra

V. ATTENUATION OF X RAYS AND GAMMA RAYS

When calculating the attenuation of these radiations, it is important to note that the attenuation equation is applicable only to monoenergetic radiation. The attenuation of a narrow beam of monoenergetic photons passing through an absorber can be described by the equation:

$$I = I_o e^{-\mu x} \tag{5}$$

where I_o represents the number of photons in the primary beam and I represents the number of photons remaining in the beam after passing through an absorber of thickness, x. The linear attenuation coefficient, μ, depends on the material of the absorber and the energy of the incident photons. Since Eq. (5) holds for a single photon energy, it is readily applicable to gamma-ray attenuation calculations. However, it can also be applied to continuous-spectrum calculations by dividing the spectrum into small energy intervals or bins, and applying the equation separately for the average energy of each bin. In this way one can calculate the effect of aluminum or copper absorbers commonly used in conjunction with x-ray machines to harden (i.e., make the average photon energy higher) the x-ray spectrum. Since μ becomes smaller as the photon energy increases, the low-energy part of the continuous spectrum is more effectively attenuated by the absorber than is the high-energy part of the spectrum. Although the resulting photon beam is reduced in intensity, it is shifted to a higher average photon energy by the selective attenuation. For an example of selective attenuation and the resulting spectral shifting, compare the curves of Figure 3.

VI. EFFECTS OF MACHINE PARAMETERS ON X-RAY SPECTRA

The other curves in Figure 3 show the effects of adding different amounts of external filtration. As more aluminum is added, the average photon energy increases, but the total number of photons in the useful beam is reduced. The curve labeled "no filtration" is the theoretical photon distribution from a thick target if it is postulated that no photons are attenuated. Of course, in actuality there is always some attenuation; some photons produced below the surface of the target are absorbed in the target, and further attenuation occurs in the glass envelope of the x-ray tube. The curve labeled "practical bremsstrahlung spectrum" approximates this inherent target and envelope attenuation under the assumption that 0.5 mm of aluminum is equivalent to this inherent filtration.

Another variable machine parameter is tube current (number of electrons impinging on the target per unit time). It is usually measured in milliamperes (mA). The effect on the x-ray spectrum of changing tube current from an arbitrary amount (C mA) to half that amount (C/2 mA), while holding the tube voltage constant at 100 kV and the filtration at 0.5 mm Al, is shown in Figure 5. The spectral shape does not change; only the number of photons delivered at each energy differs, by a factor of 2.

A third variable machine parameter is the electric potential across the x-ray tube. This potential, usually measured in kilovolts, determines the energy the electrons have when they arrive at the target. Figure 6 shows the effect of increasing the potential from 50 kV (constant potential) to 100 kV (constant potential) while keeping the current fixed and the filtration the same 0.5 mm Al . It is observed that the tungsten K characteristic x rays do not appear in the 50 kV spectrum, but are present in the 100 kV spectrum. About 69.5 keV are necessary to remove a

K electron from a tungsten atom; at 50 kV not enough energy is available.

For theoretical or idealized spectra the effect of potential on the relative number of photons produced can be expressed by the bremsstrahlung efficiency equation:

$$F = kZV \tag{6}$$

where:

F = fraction of electrons effective in producing bremsstrahlung photons,
k = a constant (about 1.2×10^{-9} in the diagnostic x-ray region),
Z = atomic number of the target material,
V = potential across the x-ray tube in volts.

In other words, when V is doubled, the number of bremsstrahlung photons produced is doubled. In actual practice, however, this 2:1 ratio does not hold. If any filtration is present (as it always is for practical x-ray tubes), the ratio is larger than for the idealized case, because the lower kV ideal spectrum contains a greater percentage of very low energy photons which are easily removed by filtration. For example, as shown in Figure 6, the ratio between the area under the 100 kV spectrum and the area under the 50 kV spectrum, each with 0.5 mm Al, is about 4:1. As the filtration is increased further, the ratio between the two areas continues to increase.

VII. OTHER SOURCES OF X RAYS

While the discussion here has been confined to the production of x rays in tubes specifically designed for this purpose, one should bear in mind that any time a source of electrons, high voltage, and a target are present, the production of x radiation is possible. Electronic tubes, such as high-voltage rectifiers and regulators, have been found to emit x rays. X rays can also be produced by supervoltage machines, such as Van de Graaff generators, betatrons, and synchrotrons. Another possible source of x rays are beta-emitting radionuclides. For this reason beta emitters are usually packaged in materials of low atomic number (such as plastic) to minimize bremsstrahlung production and to reduce the energies of the characteristic x rays produced.

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Introduction to Particle Accelerator Types and Uses

I. INTRODUCTION

Accelerators are machines which impart energy to electrically charged particles. In a strict sense, the first accelerator was the x-ray machine invented by **Konrad Roentgen** in 1895. Early x-ray machines accelerated electrons to energies less than 100 keV, and were used mainly for medical diagnostic purposes. Research scientists soon recognized the accelerators' potential as a source of high-energy particles to study nuclear reactions. It is not surprising, therefore, that most accelerator development in the past half century has been directed toward research uses. Now medicine and industry recognize and use accelerators for more practical purposes.

Today's accelerators are many and varied. In an attempt to reach higher and higher energies to dissect the nucleus, the nuclear physicist builds larger and larger accelerators. Sometimes, formidable size and attendant cost restricts use of this equipment to research. For most industrial and medical uses, however, sufficient-energy accelerators have already been obtained, and are now coming into widespread use.

The energy imparted to charged particles by accelerators can be used and are used to alter the molecular, atomic, and nuclear structure of matter. Molecular properties of such materials as plastics, resins, and transistors can be beneficially altered by accelerator beams. X rays for medical therapy or industrial radiography result from the interaction of accelerated particles and an appropriate target material. Radioactive isotopes can be created by accelerating charged particles into the nuclei of stable atoms.

II. SIMPLE ACCELERATORS

Simple accelerators can be defined as machines which accelerate particles between two electrodes by placing a potential difference between the electrodes. Simple accelerators have also been referred to as "potential drop accelerators" and "direct accelerators." There are several components common to nearly all simple accelerators. (See Figure 1.) These are: (1) a high-voltage source used as the source of energy for the system; (2) a source of charged particles to be accelerated; (3) an evacuated accelerator tube in which the particles gain their energy; and (4) a target whose purpose is either to convert the charged particle energy to x rays or act as the sample being

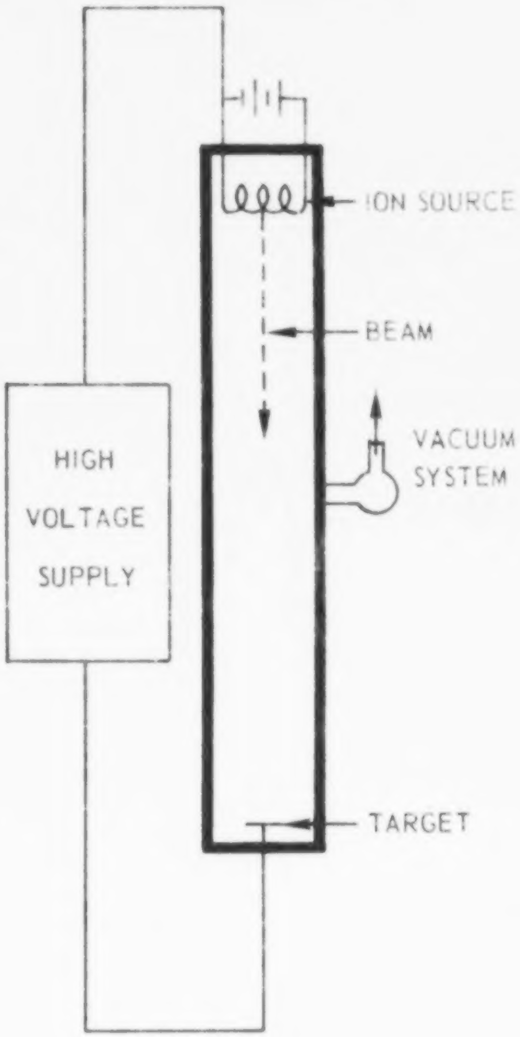


Figure 1. Basic Components of a Simple Accelerator

irradiated. The first item, high-voltage source, is used to differentiate between types of simple accelerators. In fact, most simple accelerators are named according to the type of high-voltage supply used.

Several types of simple accelerators and their uses will now be discussed.

A. Van de Graaff

The Van de Graaff accelerator—sometimes referred to as an electrostatic generator—was invented in 1931 by R. J. Van de Graaff of the Massachusetts Institute of Technology. It was designed for use in basic research. One of its principle uses is in the study of energy thresholds for nuclear reactions. Its value was soon recognized as a high-energy source of both x rays and electrons, and is used today in both medical therapy and industrial radiography. The high-voltage supply for the Van de Graaff consists of a metal dome (insulated from other parts of the accelerator), a belt of nonconducting material driven by a motor, and a source of electrical charge. (See Figure 2.)

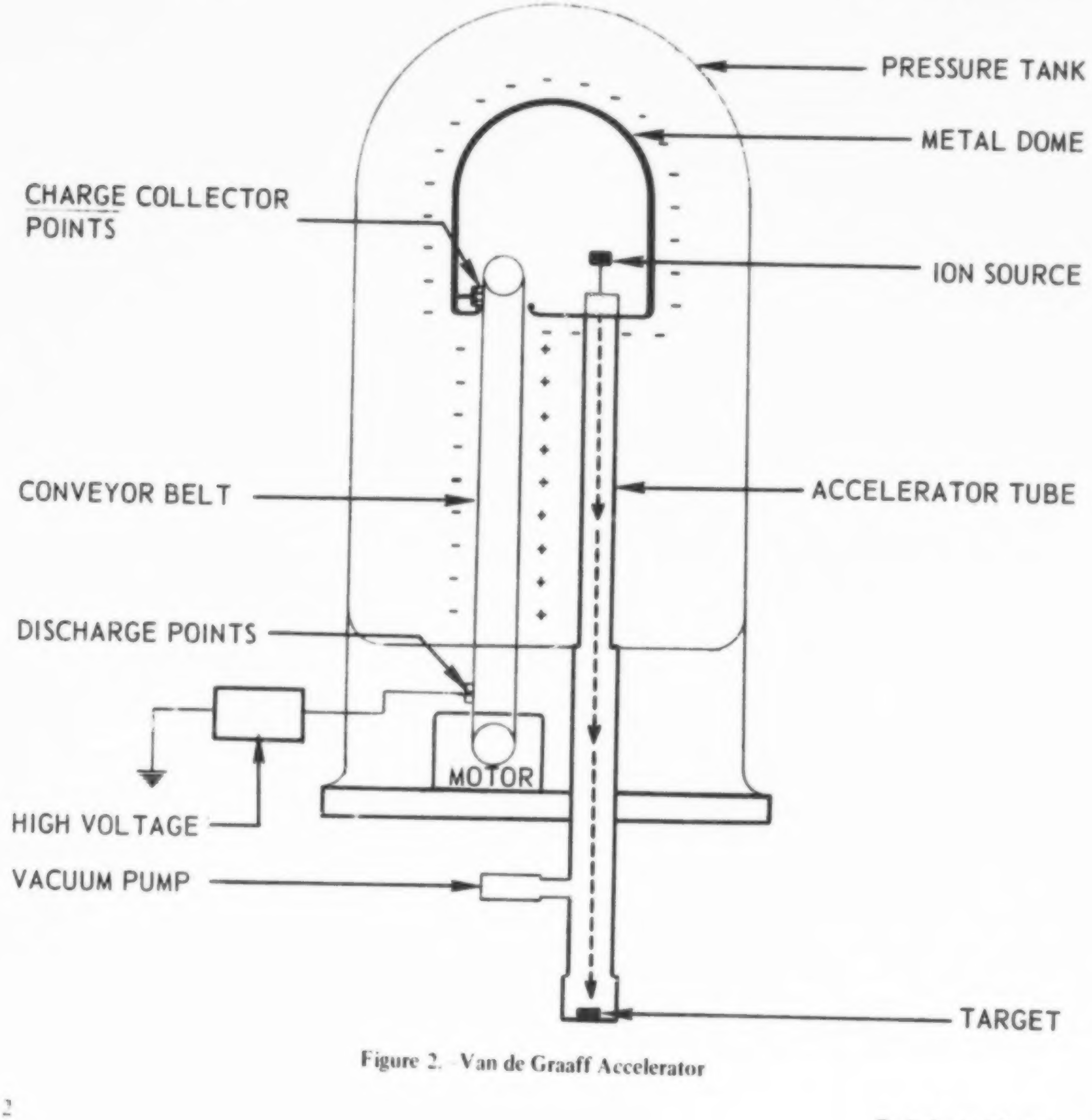

PRESSURE TANK
METAL DOME
CHARGE COLLECTOR POINTS
ION SOURCE
CONVEYOR BELT
ACCELERATOR TUBE
DISCHARGE POINTS
MOTOR
HIGH VOLTAGE
VACUUM PUMP
TARGET

Figure 2.—Van de Graaff Accelerator

The high voltage is attained in the following manner: The moving belt picks up electrical charges at the bottom of the accelerator and deposits them on the interior surface of the dome at the top of the accelerator. On reaching the interior surface of the dome, the charges will immediately move to the outer surface, where they will be uniformly distributed. This phenomenon allows for more charges to be added to the interior surface of the dome, which in turn will move to the outside surface. Because of this buildup of charge, the dome will act as the high-voltage terminal for the accelerator. Theoretically, an infinite amount of charge or high voltage could be placed on the dome if it were made large enough. However, because of insulation leakages, an upper limit of voltage will be reached when the rate of leakage equals the rate at which charge is being added to the dome. The first Van de Graaffs attained voltages of 1.5 million volts. By improved design and by enclosing the accelerator within a steel hull containing insulating gas such as freon or nitrogen under pressure voltages as high as 10 million volts have been obtained.

Depending on its use, the Van de Graaff has been used to accelerate either electrons or positive ions. For electron acceleration, the dome is made negative; for positive ions, the dome is made positive. In either case, there must be a source of these ions located at the high-voltage end of the evacuated accelerator tube. The ions are accelerated down the tube and may be allowed to pass through a thin window at its base to be used directly. If electrons are accelerated, they may strike a target (usually tungsten) at the base of the tube in order to act as a source of x rays.

In working with energies below 10 MeV, there are certain advantages to the use of the Van de Graaff: The accelerating voltage may be regulated to a high degree of accuracy—in some cases ∓ 0.1 percent. The accelerating voltage may also be varied continuously over a wide range. Another favorable feature of the Van de Graaff is its ability to reverse polarity, and thereby accelerate particles of either charge. An educational or research institution would capitalize best on these characteristics of the Van de Graaff generator. However, it still remains an excellent source of x rays in the 2-4 MeV range for use in medicine and industry.

B. Cockcroft-Walton

The first artificially-induced disintegrations of atomic nuclei by laboratory-accelerated particles were accomplished in 1932 by two English scientists, J. D. Cockcroft and E. T. S. Walton. They invented and used a high-voltage supply which now bears their names. (See Figure 3.) In essence, it takes an initial low voltage and multiplies it with a number of capacitor stages arranged in a special way. By combining rectifiers or electronic valves, the capacitors are charged in parallel and discharged in series. In this way, the initial voltage is multiplied by the number of capacitors in the machine. Insulation difficulties make the Cockcroft-Walton generator most useful in the voltage range below one million volts. The Cockcroft-Walton generator can, however, deliver moderately large currents, often several milliamperes.

A modification of the Cockcroft-Walton generator is the Dynamitron. The Dynamitron also obtains its high voltage through a series of capacitors and rectifiers, but in a much more compact manner than the Cockcroft-Walton. The Dynamitron can be readily insulated with pressurized gas. It can, therefore, attain voltages of several million volts.

Another variation of a Cockcroft-Walton generator that should be mentioned is the Marx generator. The Marx generator is essentially the same as the Cockcroft-Walton , except that it is designed to produce pulses of very short durations (sparks) rather than a continuous beam current.

Most Cockcroft-Walton's are used as a source of low energy (100-1000 keV), high-intensity positive ion beams. Specific uses include irradiation sources in industrial processes, and pre-accelerator units for injecting particles into higher energy machines.

Cockcroft-Walton generators have found increasing use in the past few years as sources of positive ion beams in the neutron generator. In the neutron generator the deuteron beam (^2H) impinges on a tritium target, thereby releasing 14 MeV neutrons at a high yield. Neutron generators have found extensive use in research and industry as inexpensive activation sources.

C. Transformers

Various types of transformers have long been used as high-voltage sources for accelerators. Basically, a transformer consists of a core (usually iron), around

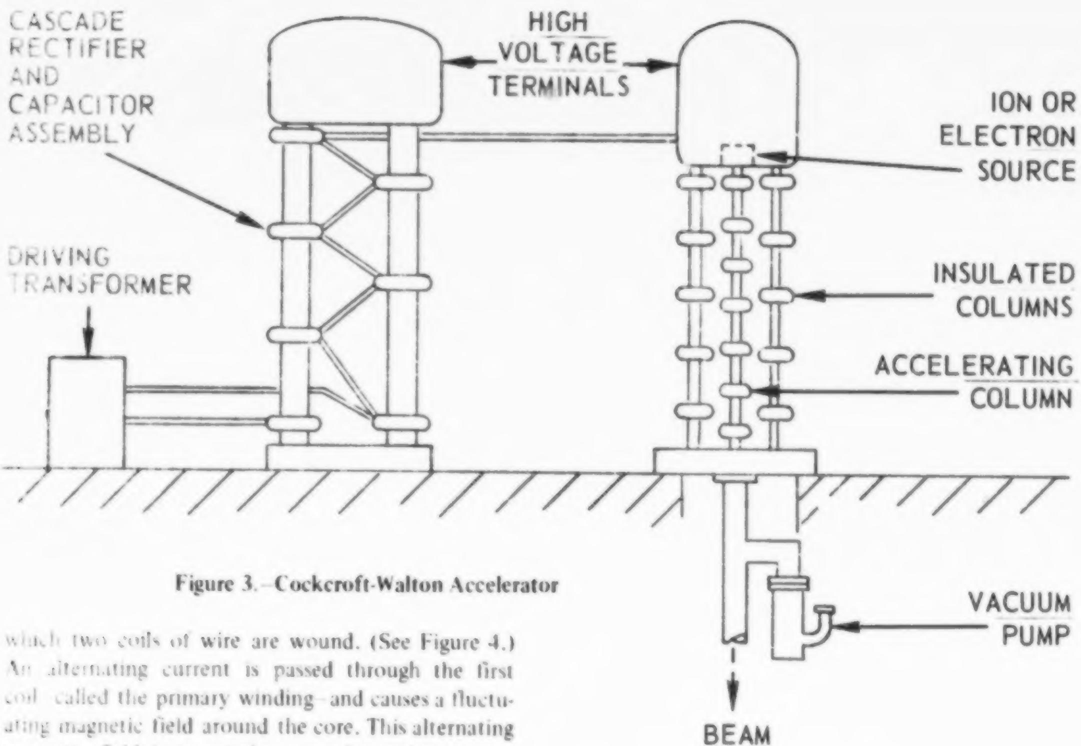


Figure 3.–Cockcroft-Walton Accelerator

which two coils of wire are wound. (See Figure 4.) An alternating current is passed through the first coil–called the primary winding–and causes a fluctuating magnetic field around the core. This alternating magnetic field in turn induces an alternating current in the second coil–called the secondary winding. The voltage in the secondary coil will be equal to the ratio of secondary coil turns over primary coil turns times the voltage in the primary coil. Thus, if the primary coil has one turn and one volt is applied to it, a secondary coil of 1000 turns will develop 1000 volts. This increase in voltage in the secondary winding is accompanied by a corresponding decrease in current. Voltages in the order of several hundred kilovolts have been developed in iron core transformers, and have been the traditional power source for x-ray machines.

With a number of refinements, scientists and engineers have developed transformers capable of obtaining several million volts. In the *resonance transformer* the accelerating tube is located within the secondary winding (see Figure 5), thus providing a more compact configuration. Resonance transformers are capable of outputs of several million volts and several milliamperes. They are used as x-ray sources in medical therapy and industrial radiography. One disadvantage of this type of transformer is that the output voltage is necessarily alternating in character. Because of this, only half of each cycle can be used for accelerating particles–the positive half for positive ions and the negative half for electrons. The resultant beam output, therefore, will be pulsed and vary in energy with the sinusoidal change of voltage during the beam pulse.

The *insulating core transformer (ICT)* on the other hand, is an accelerator transformer designed to produce an output of constant voltage and fairly high current. Its name is derived from the fact that its core is divided into segments, separated from each other. Normally, the accelerator tube is a separate unit connected to the transformer by a high-voltage cable. (See Figure 6.) Because of its high current, the ICT has found particular use in high-intensity industrial electron processing.

III. COMPLEX ACCELERATORS

In simple accelerators, the energy is delivered to the accelerating particle in one stage across one set of

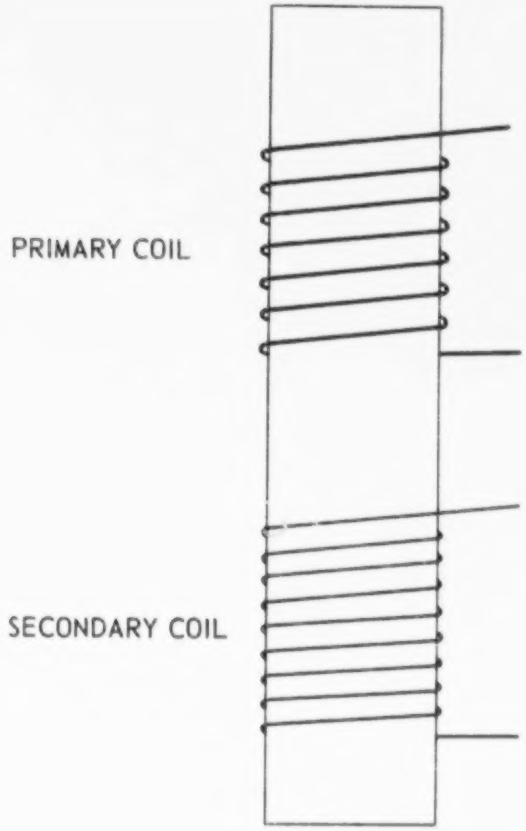


Figure 4.—Transformer

electrodes. The amount of energy that can be practically developed with this method is limited by insulation problems to something less than 12 MeV. To achieve higher energy and more penetrating particles, it is necessary to go to accelerators which deliver their energy in more than one stage, here classified as complex accelerators. The most common types of accelerators operating below 100 MeV will be briefly described.

A. Linear Accelerators

In the interest of simplicity, linear accelerators or linacs may be thought of as machines in which a series of doughnut-shaped electrodes are arranged in a straight line. (See Figure 7.) These electrodes are connected to alternating voltage sources, so tuned that as the particles leave each electrode, they are attracted to the one directly in front of it. The effect is cumulative as the particles move down the line of electrodes; by the time they reach the target, the particles may have achieved considerable energy.

In actuality, linear accelerators are more complex. Because high-frequency alternating power is needed for their operation, they cannot operate on 60-cycle electrical energy, but need the higher frequency of microwave energy. There are two basic types of linear accelerators used today. The standing-wave accelerator and the traveling-wave accelerator.

In the standing-wave design, particles are accelerated by introducing the microwave energy between gaps of a segmented accelerator tube. Standing-wave accelerators are used mainly in research installations to accelerate positive ion beams with a high current.

The traveling-wave accelerator uses microwave power in which the field is caused to move down the accelerator tube, carrying down the charged particles (usually electrons) with it. Electrons, because of their small mass, very quickly approach the speed of light in the accelerator. Therefore, most of the energy gained by electrons in high-energy accelerators is caused by an increase in relative mass rather than increased velocity.

The design of the accelerator tube, in this case called a wave guide, is the key to its operation. It is designed to keep the velocity of the wave slightly below the velocity of light, thereby permitting it to transfer energy to the electron beam. The electrons can be visualized as being trapped in the crest of a moving wave continually receiving energy from the wave in the form of increased mass as it travels down the tube. The longer the tube or wave guide, the higher the energy the electrons can obtain. The microwave energy is supplied by klystron tubes similar to those employed in radar transmitters. The Stanford Linear Research Accelerator at Palo Alto, California, is the largest of its kind in the world. It is two miles long and can attain electron energies in excess of 20 GeV.

Much smaller traveling-wave accelerators have found practical use for electron and x-ray therapy (see Figure 8), electron beam processing, and supervoltage radiography. A typical linear accelerator available for x-ray therapy today would operate at an energy of about 6 MeV. Like all linear accelerators, it

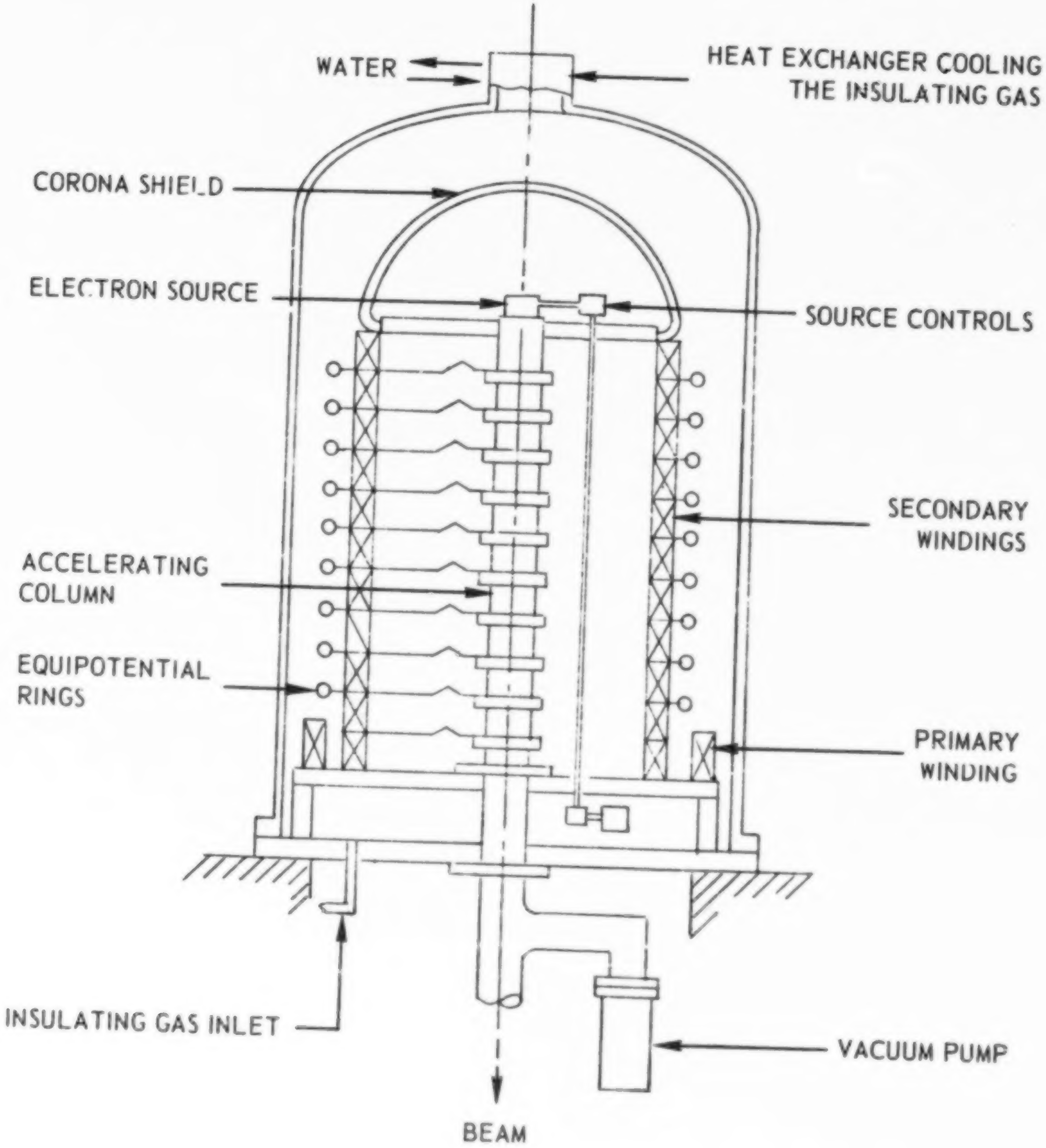

WATER
HEAT EXCHANGER COOLING THE INSULATING GAS
CORONA SHIELD
ELECTRON SOURCE
SOURCE CONTROLS
SECONDARY WINDINGS
ACCELERATING COLUMN
EQUIPOTENTIAL RINGS
PRIMARY WINDING
INSULATING GAS INLET
VACUUM PUMP
BEAM

Figure 5. Resonant Transformer Accelerator

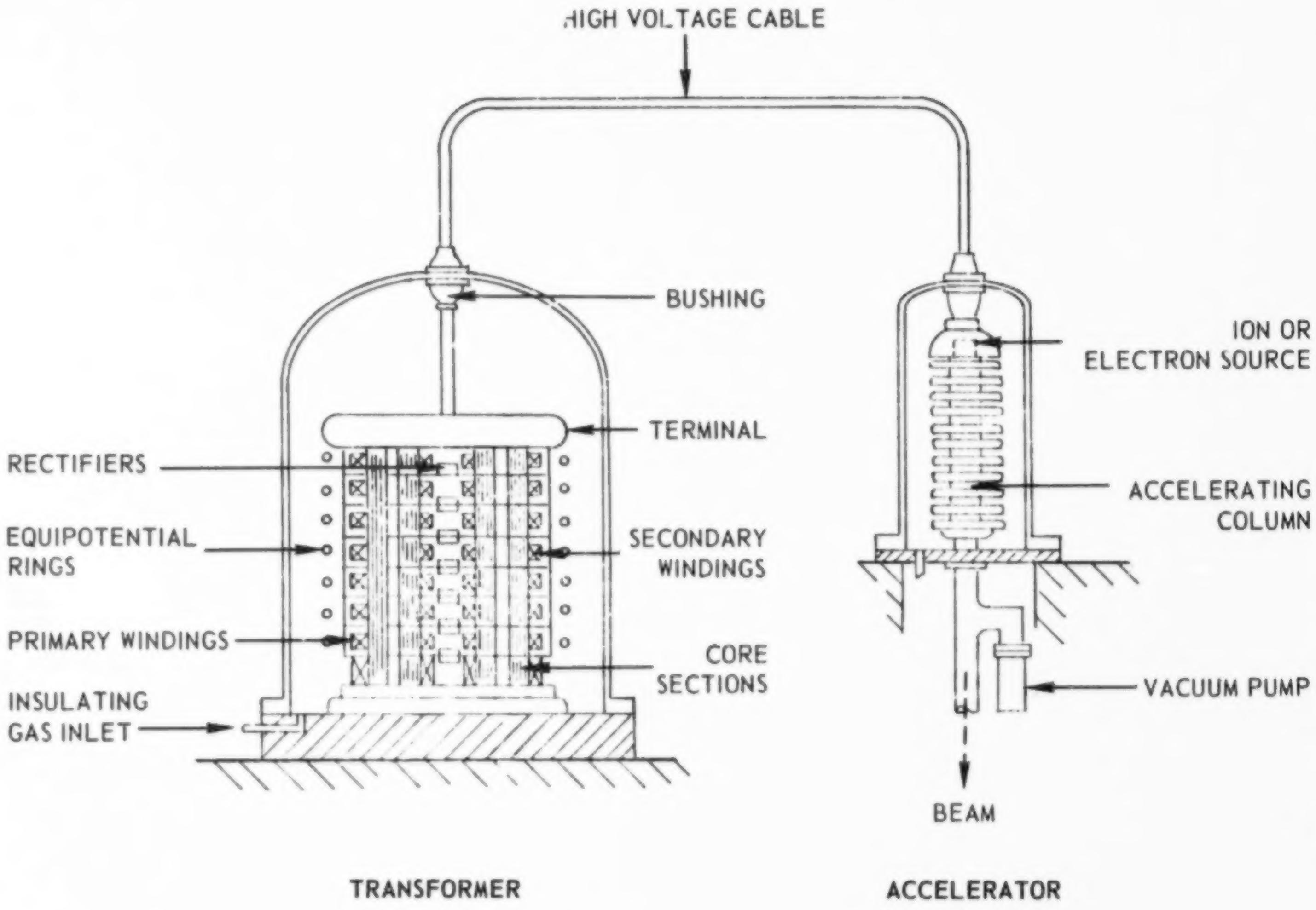

HIGH VOLTAGE CABLE
BUSHING
ION OR ELECTRON SOURCE
TERMINAL
RECTIFIERS
ACCELERATING COLUMN
EQUIPOTENTIAL RINGS
SECONDARY WINDINGS
PRIMARY WINDINGS
CORE SECTIONS
VACUUM PUMP
INSULATING GAS INLET
BEAM
TRANSFORMER
ACCELERATOR

Figure 6.—Insulating Core Transformer Accelerator

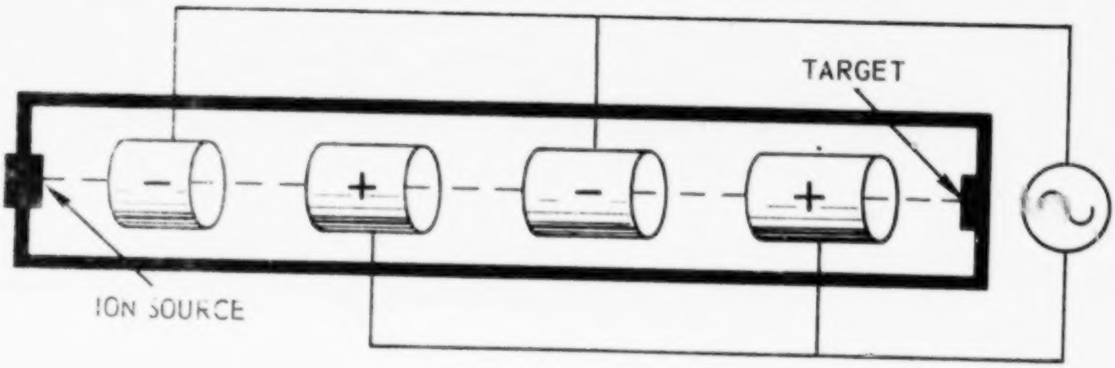


Figure 7.– Linear Accelerator

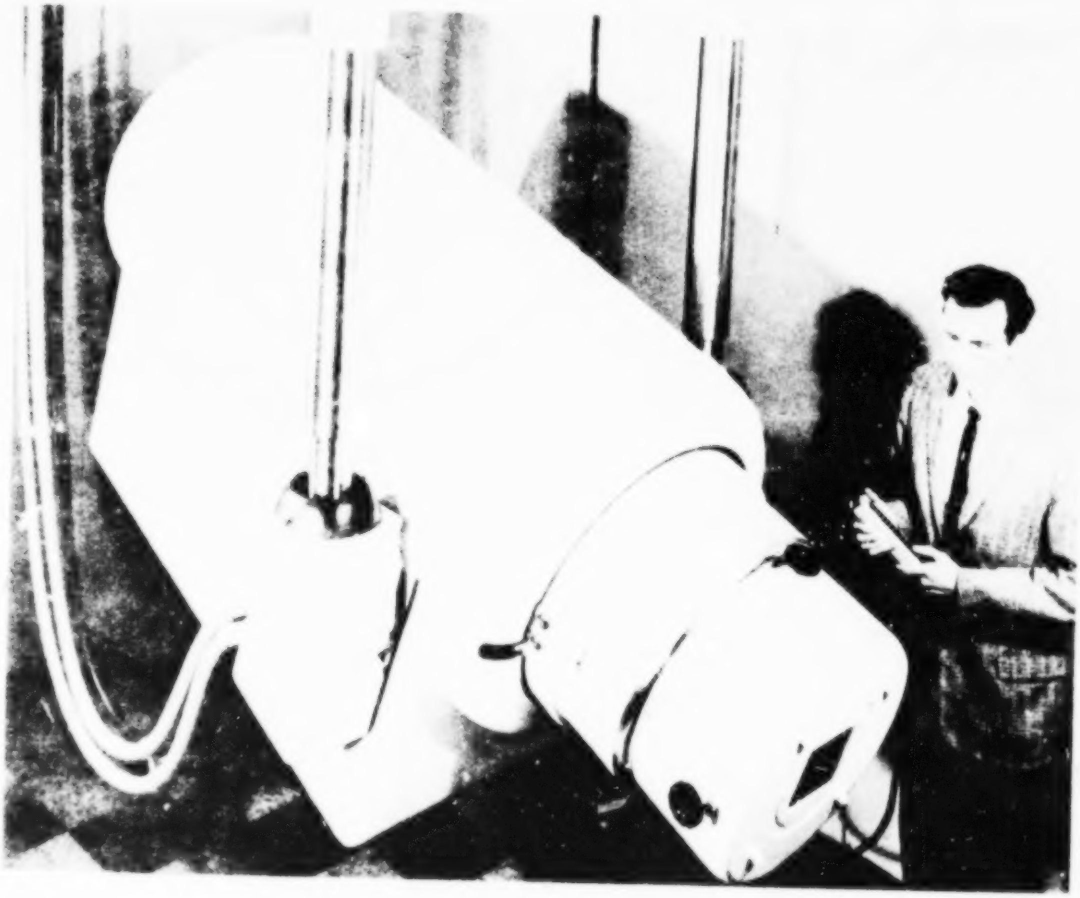


Figure 8.–Traveling Wave Linear Accelerator
(Courtesy of High Voltage Engineering Corp.)

would have a pulsed beam. The linear accelerator enjoys a big advantage over its competitors in that it does not need an insulation tank to reach MeV energies. This allows for a very compact unit in the 2 to 6 MeV range.

B. Cyclotrons

In principle, the cyclotron is very similar to the linac. The accelerating particles receive their energy in many stages from a high-frequency microwave source. The main difference is in the path the particle follows. In the cyclotron, the particle follows a circular path; in the linear accelerator, the particle follows a straight line. Particles in a cyclotron are held in a circular orbit by means of an intense magnetic field. The particles start out at the center of the cyclotron in a gap between two hollow semicircular electrodes called "dees." In the gap, they receive an initial acceleration. They then enter a dee, where they coast in a circular path until they reenter the gap to receive a further push. As their energies increase, their path radius increases, and the particles spiral outward. When the particles reach the outside of the cyclotron, they are directed to a target. Several thousand crossings may occur before this happens and particles may accumulate energies of many MeV. A schematic drawing of a vertical and horizontal section of a cyclotron is shown in Figure 9.

The upper limit of energy attainable with a cyclotron is set by the relativistic increase in mass experienced by matter as it approaches the velocity of light. This relativistic mass increase makes it very difficult to synchronize the gap pushes properly. The limiting energy will depend on the mass of the accelerating particle. For protons, this is about 25 MeV. For heavier particles, such as deuterons or alpha particles, it is somewhat higher. Because of their very light mass, electrons can only be brought up to about 10 keV in a cyclotron. This explains why electrons are not accelerated in cyclotrons.

Cyclotrons have traditionally been used as a source of high-energy positive ions in research installations. Before the advent of reactors, they were the biggest producers of radioisotopes. In recent years, there has been renewed interest in the use of cyclotrons for the generation of short-lived isotopes close to where they will be used. Medical installations are particularly interested in obtaining short-lived isotopes for diagnostic and therapeutic purposes.

C. Race Track Accelerators

In order to make advances in high-energy physics, it has been necessary to modify the cyclotron to overcome the relativity problem mentioned. Successive modifications have enabled scientists to achieve higher and higher energies and, therefore, larger and larger machines. Because of the money involved in building and operating these big machines, it is not surprising that each one of these machines is almost unique in itself. Basically, these machines overcome the relativistic mass problem by either varying the magnetic field or the power supply frequency as the particles are accelerated. It is beyond the scope of this presentation to describe the actual mechanisms of their operation.

More than a dozen of these high-energy machines have been built or are being built throughout the world. The largest is the proton synchrotron under construction at Weston, Illinois, which will achieve proton energies in excess of 200 GeV. Looking at the design of this machine, it is easy to see why it is classed as a "race track accelerator". It will have a diameter of approximately 1.25 miles.

D. Betatron

The betatron is a special type of circular accelerator which accelerates electrons to high energies. It consists of an evacuated donut-shaped tube located between the poles of a large magnet. (See Figure 10.) Electrons released within the tube are accelerated by an alternating magnetic field. In the betatron, the magnetic field acts both as a power supply for accelerating the electrons and as a positioner to hold the electrons in a circular path. Betatrons generate a pulsed beam, which may attain energies of 100 MeV. They have found practical application as x-ray sources for radiographing heavy castings and for high-energy medical therapy.

IV. SUMMARY

There are many different types of accelerators used in many different ways today. In this presentation, emphasis has been on the lower-energy machines. The use of these small accelerators for medical and industrial purposes has significantly increased in recent years.

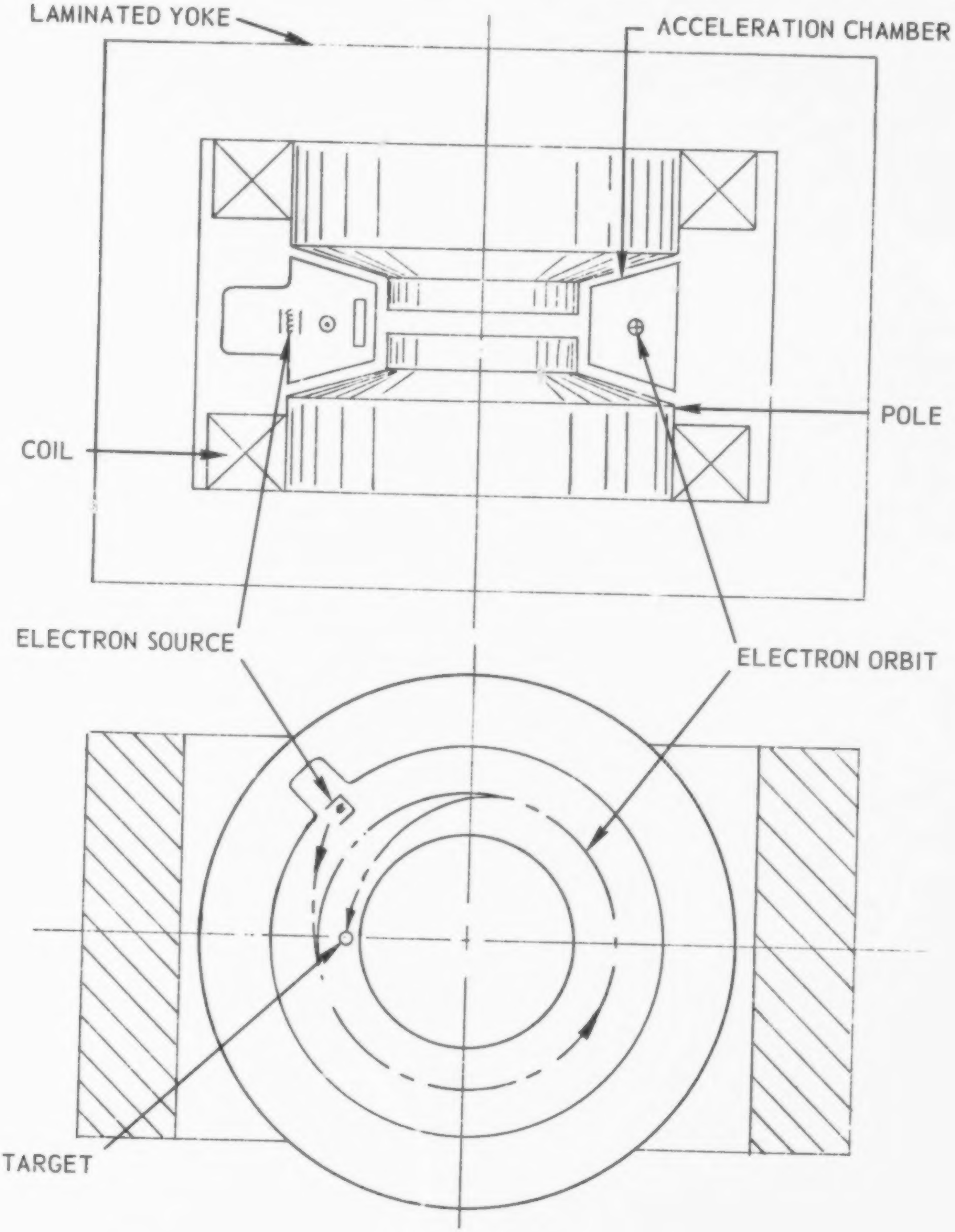
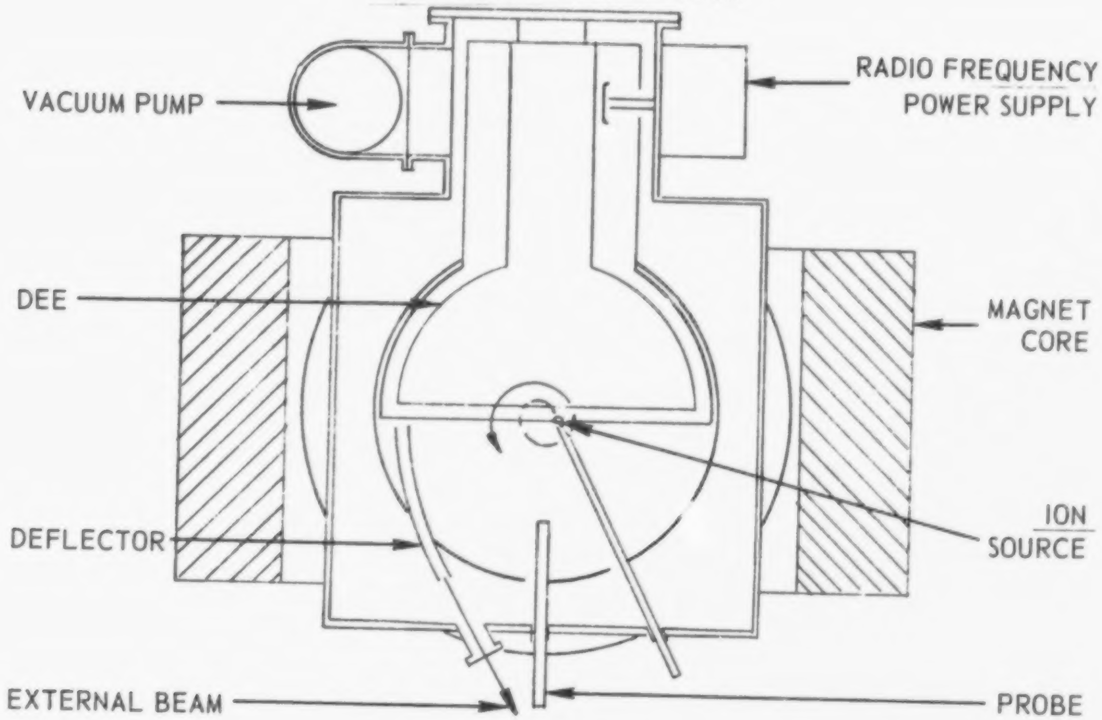

LAMINATED YOKE
ACCELERATION CHAMBER
POLE
COIL
ELECTRON SOURCE
ELECTRON ORBIT
TARGET

Figure 9. - Cyclotron

HORIZONTAL CROSS SECTION


VACUUM PUMP
RADIO FREQUENCY POWER SUPPLY
DEE
MAGNET CORE
DEFLECTOR
ION SOURCE
EXTERNAL BEAM
PROBE

VERTICAL CROSS SECTION

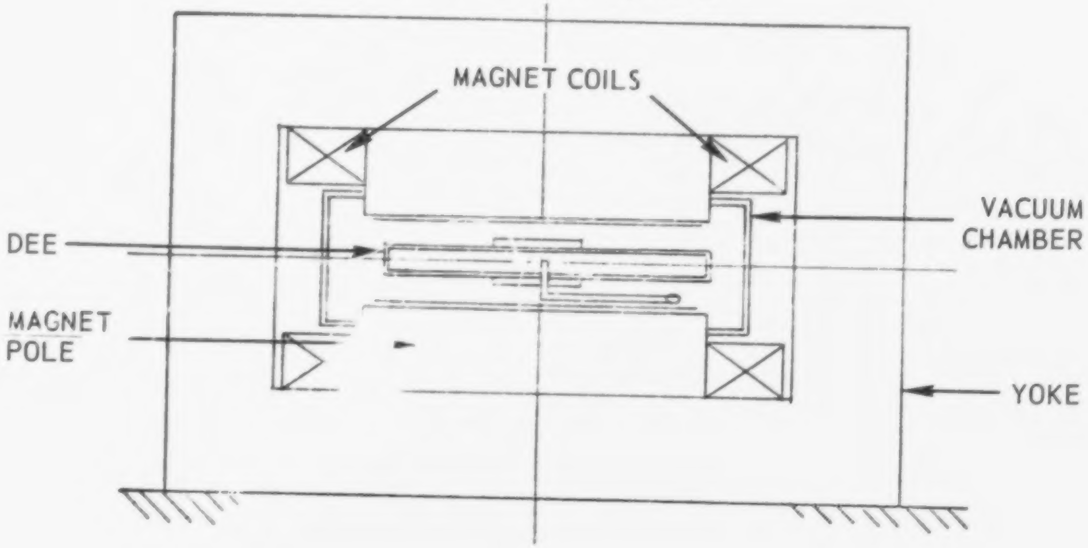

MAGNET COILS
VACUUM CHAMBER
DEE
MAGNET POLE
YOKE

Figure 10.–Schematic of a Betatron

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Nuclear Reactions with Charged Particles and Photons

I. INTRODUCTION

A nuclide may be changed into another nuclide (a phenomenon known as transmutation) by bombarding it with charged particles or photons. This process is important since the product nucleus of the reaction may be radioactive and the secondary particles given off can cause other reactions.

II. THEORY OF NUCLEAR REACTIONS

The initial, intermediate, and final steps of a typical nuclear reaction involve five components, as shown below:

$$\begin{bmatrix}\text{Target}\\ \text{Nucleus}\end{bmatrix} + \begin{bmatrix}\text{Bombarding}\\ \text{Particle}\end{bmatrix} \rightarrow \begin{bmatrix}\text{Compound}\\ \text{Nucleus}\end{bmatrix} \rightarrow \begin{bmatrix}\text{Product}\\ \text{Nucleus}\end{bmatrix} + \begin{bmatrix}\text{Product}\\ \text{Particle}\end{bmatrix}$$

A. Components of the Reaction

The components of the reaction are described as follows:

1. Examples of a bombarding particle or projectile are an alpha or beta particle, a gamma photon, a proton, or a deuteron. Sources of these may be fission, fusion, radioactive decay, particle accelerators, or transmutations.

2. The target nucleus may be any nucleus ranging from that in the lightest atom, hydrogen, to that in one of the heaviest, uranium.

3. The product nucleus is the nuclear species formed as a result of the breakdown of the compound nucleus.

4. The product particle may be any of the subatomic particles. More than one product particle may be given off.

5. The compound nucleus, containing the target nucleus plus the bombarding particle, exists for only 10^{-12} seconds or less.

a. During this lifetime the excess energy of the particle is distributed throughout the nucleus and the incident particle loses its identity.

b. The method of breakdown of the compound nucleus is dependent only on the excess energy contained, not on the method of formation.

c. Disintegration of the compound nucleus differs from radioactive decay in two ways:

(1) the lifetime of the compound nucleus is always extremely short.

(2) protons are not converted into neutrons and neutrons are not converted into protons during the process of change.

B. Formation of Compound Nucleus

The extent of nuclear interaction depends on the nature of the projectile, its energy, and the material being bombarded:

1. Before positively charged particles can interact with a nucleus, they must overcome a potential energy barrier--the electrostatic force of repulsion. This force is proportional to the charge on both the target and projectile; the greater the charge, the greater the energy necessary to overcome the barrier. For this reason, reactions occur most easily between projectile particles and targets of low atomic number.

2. Beta particles and photons interact primarily with electrons and are thus more effective in bombarding a nucleus having few orbital electrons, i.e., of low atomic number. Since these projectiles can offer little or no mass-energy to the compound nucleus, they must have kinetic energy greater than the binding energy of a nucleon (proton or neutron) before a particle can be ejected from

the target nucleus. For this reason transmutation by beta particles and photons of energies less than about 5 MeV will occur only in special cases. At high energies, these projectiles can cause many transmutations.

C. Conditions for Decay

Three laws may be applied to most nuclear reactions involving energy changes of less than a few MeV: (1) conservation of mass number, (2) conservation of atomic number, and (3) conservation of energy. These laws simplify writing of reactions and predicting possible products.

1. For the mass number to be conserved, the total number of nucleons included in the system before the reaction must equal the total number after the reaction.

2. For the atomic number to be conserved, the number of protons in the system must remain constant. Thus, no protons can be changed into neutrons (as in electron capture) nor can any neutrons be changed into protons (as in beta decay). The total number of neutrons also remains constant.

3. Conservation of energy specifies that the total energy on one side of the equation must equal that on the other. Since mass may be converted into energy in a nuclear reaction, the factor being conserved is usually designated as:

$$\begin{bmatrix}\text{Kinetic}\\ \text{Energy}\end{bmatrix} + \begin{bmatrix}\text{Mass}\\ \text{Energy}\end{bmatrix}$$

where the equivalence of mass and energy is computed by the relationship:

1 atomic mass unit = 931 MeV.

The following is an example of the application of the principle of conservation of total energy:

$$^{14}_{7}\text{N} + {}^{4}_{2}\text{He} \leftarrow \left[{}^{18}_{9}\text{F}\right] \leftarrow {}^{17}_{8}\text{O} + {}^{1}_{1}\text{H}$$

$$14.00307 + 4.00260 \leftarrow 16.99914 + 1.00782$$

$$\Delta m = \text{Mass}_{\text{reactants}} - \text{Mass}_{\text{products}}$$

$$\Delta m = 18.00567 - 18.00696 = -0.00129$$

$$\Delta E = 931\ (0.00129) = 1.20\ \text{MeV}$$

Since there is a gain in mass of the reactants, 1.20 MeV of excess energy must be supplied to produce this reaction. This is called an endoergic reaction. If the mass of the reactants is more than the mass of the products, the reaction is exoergic, or energy is released. If atomic weights of the target and product nuclei are used, such calculations will be accurate only if atomic weights are also used for the bombarding and product particles. In this way, the total mass of the electrons on each side of the equation balances out.

III. EXPRESSING NUCLEAR REACTIONS

Several shorthand methods can be used to indicate nuclear reactions.

A. The reaction,

$$^{23}_{11}\text{Na} + {}^{1}_{1}\text{H} \leftarrow \left[{}^{24}_{12}\text{Mg}\right] \leftarrow {}^{23}_{12}\text{Mg} + {}^{1}_{0}\text{n}$$

can also be written:

$$^{23}\text{Na}\ (\text{p,n})^{23}\text{Mg}$$

B. Since only three of the four products and reactants of a reaction are necessary to balance the equation, this reaction could also be designated by: ^{23}Na (p, n).

IV. TYPICAL REACTIONS

Depending on the energy of the bombarding particle and its probability of interaction with a target nucleus, nuclear reactions can be classified in a number of ways. These reactions are general and cannot be assumed to occur with any particular target nucleus.

A. Proton-Induced Reactions

1. (p,γ) and (p, n) reactions are common with targets of low atomic number. A typical example is:

$$^{27}_{13}\mathrm{Al} + {}^{1}_{1}\mathrm{H} \leftarrow {}^{28}_{14}\mathrm{Si} + \gamma$$

2. High energy protons can cause (p, 2n) and (p, pn) reactions. Three or four particles may be emitted if the proton energy is several hundred MeV.

B. Deutron-Induced Reactions

1. The most common reactions are the (d,p) and (d, n) reactions:

$$^{130}_{52}\mathrm{Te} + {}^{2}_{1}\mathrm{H} \leftarrow {}^{131}_{53}\mathrm{I} + {}^{1}_{0}\mathrm{n}$$

2. At high energies, (d,2n) or (d,α) reactions may also occur.

C. Alpha-Induced Reactions

1. A number of reactions of the (α, n) type may occur if the target nucleus is an element of fairly low atomic number. The reaction

$$^{9}_{4}\mathrm{Be} + {}^{4}_{2}\mathrm{He} \leftarrow {}^{12}_{6}\mathrm{C} + {}^{1}_{0}\mathrm{n}$$

led to the discovery of the neutron by Chadwick in 1932. This reaction is now a common laboratory source of neutrons, the alpha particles being supplied by one of the natural radionuclides such as polonium.

2. Multiple particle emissions such as (α,2n), (α,3n), and (α,np) will occur when alpha particles of very high energy are used as projectiles.

D. Electron (Beta particle) - Induced Reactions

1. An example of a reaction of this type is:

$$^{9}_{4}\mathrm{Be} + {}^{0}_{-1}\mathrm{e} \leftarrow {}^{8}_{4}\mathrm{Be} + {}^{1}_{0}\mathrm{n} + {}^{0}_{-1}\mathrm{e}$$

It is interesting to note that the electron simply provides energy for the reaction while passing through or very near the target nucleus.

2. A number of nuclear transmutations (or excitations) are possible with high energy electrons.

E. Photon-Induced Reactions

1. The following are two observed cases of (γ, n) reactions, often called "photo-disintegration":

$$^{2}_{1}\mathrm{H} + \gamma \leftarrow {}^{1}_{1}\mathrm{H} + {}^{1}_{0}\mathrm{n}$$

and

$$^{9}_{4}\mathrm{Be} + \gamma \leftarrow {}^{8}_{4}\mathrm{Be} + {}^{1}_{0}\mathrm{n}$$

The first reaction was used to compute the mass of the neutron by Goldhaber and Chadwick in 1934.

2. Many additional transmutations by photons are possible at higher energies (of the order of 20-330 MeV).

V. SUMMARY

Whenever charged particles and photons are able to penetrate the potential energy barrier and interact with a nucleus, transmutations may be caused. These usually result in artificial radioactivity. The bombarding projectile can be a proton, deuteron, alpha particle, electron, or γ photon. Such bombarding particles may originate from other transmutations, radioactive decay, fission, fusion, or particle accelerators. As a result of the interaction of the projectile and target, a compound nucleus is formed, exists for an instant, and then separates into a product particle and product nucleus. Three laws govern these reactions: (1) conservation of mass number, (2) conservation of atomic number, and (3) conservation of total energy.

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Neutrons

I. INTRODUCTION

The existence of a neutral particle equal to the combination of a proton and an electron was suggested in 1920. Such a particle was thought to possess some novel properties and possibly be an important unit in atomic structure. Unsuccessful attempts to detect such a particle were made for many years.

Between 1930-1932, many experiments showed bombardment of various light elements with alpha particles yielded a highly penetrating radiation. This was thought to be gamma rays of unusually high energy. Further studies, however, revealed inconsistencies of their interactions on this basis.

In 1932, James Chadwick resolved these difficulties by demonstrating that bombardment of beryllium by alpha particles produced the particles now known as neutrons. The nuclear reaction taking place was described as:

$$^{9}_{4}\text{Be} + ^{4}_{2}\text{He} \rightarrow ^{12}_{6}\text{C} + ^{1}_{0}\text{n}\ .$$

II. PROPERTIES

The neutron has a mass number of 1, an atomic number (nuclear charge) of 0, and an atomic mass of 1.008665 amu. Because it has no electrical charge, the neutron can penetrate relatively easily into a nucleus.

Free neutrons are unstable (radioactive) and disintegrate by beta emission with a half-life of approximately 13 minutes. The resultant decay product is a proton.

Neutron radiations are classified on the basis of their kinetic energies. The most probable velocity of free neutrons in various substances at ordinary room temperature is approximately 2200 meters per second. Their kinetic energy may be calculated from the equation

$$E = \tfrac{1}{2}\, mv^2$$

where:

m = neutron mass in grams
v = neutron velocity in cm/sec.

Substituting:

$$E = \tfrac{1}{2}\,(1.66 \times 10^{-24}\,\text{gm})\,(2.2 \times 10^{5}\ \text{cm/sec})^2$$

$$= 4.0 \times 10^{-14}\ \text{ergs}$$

$$= 0.025\ \text{eV}.$$

Neutrons with this average kinetic energy at 20° C. are called thermal neutrons. Epithermal, or slow, neutrons range in energy from 0.025 eV to 100 eV. Intermediate neutrons range in energy from 100 eV to 10^5 eV. Fast neutrons possess energies from 10^5 eV to 20 MeV. Relativistic neutrons have energies in excess of 20 MeV.

Classification of neutrons according to kinetic energy is important from two standpoints: (1) the interaction of neutrons with the nuclei of atoms differs with the neutron energy; and (2) the methods of producing, detecting, and shielding against the various classes of neutrons are different.

Detection of neutrons is relatively difficult, due to lack of ionization along their paths; negligible response to externally applied electric, magnetic or gravitational fields; and the fact that they interact only with atomic nuclei, which are exceedingly small.

III. SOURCES

Nuclear reactions are the only source of neutrons. Such reactions include:

1. Those induced by alpha particles. For example, the previously discussed reaction:

$$^{9}_{4}\text{Be} + ^{4}_{2}\text{He} \rightarrow ^{12}_{6}\text{C} + ^{1}_{0}\text{n}\ .$$

It will be recalled that this reaction led to the discovery of the neutron by Chadwick in 1932. It is also an important source of neutrons for initiating the fission reaction during startup of nuclear reactors.

2. Those induced by high-energy gamma photons, such as are yielded by ^{24}Na and ^{124}Sb. Light elements are the preferred target nuclei for reactions of this type. Typical examples include:

$$^{2}_{1}H + \gamma \rightarrow {}^{1}_{1}H + {}^{1}_{0}n$$

and

$$^{9}_{4}Be + \gamma \rightarrow {}^{8}_{4}Be + {}^{1}_{0}n.$$

The latter reaction may be a source of neutrons in reactors employing beryllium reflectors.

3. Those initiated by high-energy charged particles, such as may be obtained in particle accelerators. For example, a commercial source of relatively large neutron intensities has been developed using a reaction involving deuterons of only a few hundred keV energy,

$$^{3}_{1}H + {}^{2}_{1}H \rightarrow {}^{4}_{2}He + {}^{1}_{0}n.$$

Other suitable targets for this reaction are ^{2}H, ^{3}Li, and ^{4}Be.

Proton bombardment is also used to produce neutrons, as:

$$^{7}_{3}Li + {}^{1}_{1}H \rightarrow {}^{7}_{4}Be + {}^{1}_{0}n.$$

4. Those initiated by neutrons themselves--for example, the fission process.

IV. REACTIONS

Reactions may be considered from several view points. In the following paragraphs we will consider (a) general types, (b) reactions in biological injury, and (c) reactions useful in detection and measurement.

A. General Types

Capture—Radioactive capture with gamma emission is the most common type of reaction for thermal and slow neutrons. The (n,γ) reaction often results in product nuclei which are radioactive; for example:

$$^{59}_{27}Co + {}^{1}_{0}n \rightarrow {}^{60}_{27}Co + \gamma\ .$$

After capture of a neutron, the compound nucleus can sometimes decay by the emission of a charged particle. Of the possible reactions, (n,p) leading to the emission of a proton on neutron bombardment and (n,α) leading to the emission of an α particle, are the most frequently encountered. Reactions initiated by thermal neutrons are ^{14}N(n,p)^{14}C and ^{10}B(n,α)^{7}Li. A (n,p) reaction using high-energy 14 MeV neutrons is ^{16}O(n,p)^{16}N. This reaction is used to determine, in minutes, the oxygen content of certain metals and alloys.

Fission—A reaction which may follow the capture of a neutron is the fissioning of the excited nucleus into two fragments with the emission of neutrons. Nuclei which undergo fission after thermal neutron capture are ^{233}U, ^{235}U, and ^{241}Am. Those which fission with fast neutrons are ^{238}U and ^{232}Th.

Neutron Scattering—Scattering of neutrons in elastic or inelastic collisions is the principal mechanism of moderation or slowing down of fast neutrons to thermal energies.

B. Reactions in Biological Injury

Fast neutrons lose energy in soft tissue mainly by repeated scatterings with hydrogen nuclei. The hydrogen nuclei are themselves scattered in the process and become recoil protons.

Slow neutrons are captured in soft tissue and release energy by two principal mechanisms:

$$^{1}_{0}n + {}^{1}_{1}H \rightarrow {}^{2}_{1}H + \gamma(2.2\,MeV)$$

and

$$^{1}_{0}n + {}^{14}_{7}N \rightarrow {}^{14}_{6}C + p\ (0.63\ MeV).$$

C. Reactions Useful in Detection and Measurement

Neutrons produce very little direct ionization. Ordinarily they will not be readily detected by G-M counters or other ionization chambers. Such instruments can be adapted to respond to neutron radiation through certain secondary effects. The two techniques most often employed either use the charged particles produced by neutron interactions with various substances introduced into the counter, or use the recoil nuclei of light elements which undergo collisions with neutrons.

Slow neutron detectors generally use neutron reactions involving boron. For example, neutrons interacting with ^{10}B cause the reaction:

$$^{1}_{0}n + ^{10}_{5}B \rightarrow ^{7}_{3}Li + ^{4}_{2}He\ .$$

Detection of the alpha particles indicates the presence of slow neutrons. The boron may be introduced into the counting chamber either as a gas (BF_3) or as a solid liner for the chamber's inner walls.

Slow neutron detectors are also constructed using ^{235}U as an inner liner for the chamber walls. Slow neutrons cause fission of the uranium and are detected through ionization caused by the resultant fission fragments.

Fast neutrons are often measured by a detector filled with methane gas (CH_4) or lined with some other hydrogen-rich material such as paraffin. Fast neutrons striking the light nuclei cause protons to be ejected. This produces considerable ionization. Fast neutron detectors also use ^{238}U as an inner liner for the chamber walls, since it can be fissioned only by fast neutrons.

Scintillation detectors and photographic film will record neutrons with fair to good efficiency if loaded with materials which react readily with neutrons to yield secondary ionizing particles.

V. USES

Neutron uses can be illustrated by, but not limited to, (a) isotope production, (b) activation analysis, and (c) medical therapy. These are briefly discussed in the following subsections.

A. Isotope Production

As mentioned previously, the most common thermal neutron reaction is (n, γ). This reaction has been observed in nearly all elements and is an important source of many reactor-produced radionuclides. A typical example is ^{24}Na, which is produced by the reaction:

$$^{23}_{11}Na + ^{1}_{0}n \rightarrow ^{24}_{11}Na + \gamma\ .$$

A few useful radionuclides result from special reactions, for example: $^{14}N(n,p)^{14}C$, $^{32}S(n,p)^{32}P$, and $^{6}Li(n,a)^{3}H$.

B. Activation Analysis

Qualitative and quantitative chemical analyses of minute traces of elements can be made by exposing a sample to neutrons and identifying the contents by the characteristics of the radionuclides formed. Sensitivities for most elements are in the range of 10^{-16} to 10^{-11} grams. Concentrations as low as a few thousandths of a part per million can be measured.

Neutron activation analysis with gamma spectroscopy can be very fast. Moreover, although it is particularly suited for the measurement of trace quantities, it can be used for higher concentrations. And because much of this instrumentation can be automated, it can be used in the process-control field.

It is used in: agriculture, for detecting pesticide residues on crops; metallurgy, for measuring trace impurities; geology, for analyzing minerals, oil well explorations, etc.; and electronics, for measuring impurities (arsenic and copper) in silicon semiconductors.

C. Medical Therapy

Tumors may be medically treated by incorporating within them certain elements possessing large neutron-capture efficiencies. Subsequent neutron irradiation of the affected area results in the emission of energetic particles, causing heavy localized irradiation. Elements used include boron, lithium, and uranium.

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Fission and Fusion

I. INTRODUCTION

Soon after the discovery of radioactivity, physicists began to conjecture about the enormous energy apparently stored within the atom. Later, when the nuclear theory of the atom was developed, it was realized that the energy was stored in the nucleus; however, it was not known how this energy could be released.

In the course of his studies on the theory of relativity, Albert Einstein showed that mass and energy were, in a sense, equivalent. Consequently, energy would be liberated in any process associated with a net decrease in mass.

There are many nuclear reactions in which mass is converted to energy, but usually the energy required to initiate the reaction is greater than or nearly equal to the energy released. It is now known that there are two ways in which nuclear energy can be released without expending greater energy. One is by splitting the heavier nucleus into two roughly equal parts; this is called fission. The other is by combining some of the lighter nuclei by a process called fusion.

Nuclear fission was discovered in 1939; within 6 years this energy was utilized in the "atomic" bomb. Since that time, the energy of controlled nuclear fission has been converted into useful heat in nuclear reactors.

Nuclear fusion was demonstrated as a great source of energy by the thermonuclear bombs around 1952. When controlled and sustained nuclear fusion is accomplished, a nearly unlimited source of energy will be available since the fuel of nuclear fusion is deuterium which is abundant in and easily extracted from water.

II. FISSION

The various aspects of the fission process will be discussed in this section.

A. Types of Fission Reactions

Fission can be caused by the bombardment of various target nuclei with neutrons, protons, and alpha and other particles. Fission was discovered by bombarding uranium with neutrons. Later experimental evidence proved that uranium-235 undergoes fission when bombarded with neutrons of energies as low as 0.03 eV. It was found that uranium-238 and thorium-232 will undergo fission only if the neutrons have an energy of 1 MeV or greater. The fission of artificially-produced isotopes was also investigated and it was found that uranium-233 and plutonium-239 undergo fission upon interaction with neutrons of low energy (0.03 eV).

These reactions are useful because the energy output exceeds the energy input. Other fission processes may not be good energy sources but are studied to learn about the structure of the nucleus. Thus in 1947, fission of bismuth, lead, thallium, mercury, gold, platinum, and tantalum was achieved through bombardment with high-energy neutrons, protons, deuterons, and alpha particles. Fission has even been induced by bombardment with gamma rays; this process is known as photofission.

B. Probability of Nuclear Fission

The probability of a nuclear fission occurring depends on the nuclide being bombarded, and the type and energy of the bombarding particle. The probability of reaction is expressed as a "cross section." For fission reactions the probability is called the fission cross section, σ_f.

The fission cross section of uranium-233, uranium-235, and plutonium-239 with low energy (thermal) neutrons is relatively large. The fission cross section of uranium-238 and thorium-232 with thermal neutrons is zero but the cross section becomes significant when the neutron energy is above 1 MeV.

C. Mechanism of Uranium-235 Fission

If a uranium-235 nucleus is fissioned by a thermal neutron, it will usually split into two nuclei which are known as primary fission fragments (or products). Prompt fission gammas are also given off with energies as high as 10 MeV. Within 10^{-12} seconds after the fission, the primary fission fragments emit from zero to five neutrons called prompt neutrons.

There are more than 40 different ways that the nucleus might split, giving rise to more than 80 possible fragments. The fragments are usually radioactive and their daughter products are considered fission products. Usually the primary fission fragment's decay chain has three stages. Thus, one might expect any of 250 different radionuclides as a result of uranium-235 fission. The relative probability of any given primary fission fragment being formed is illustrated in Figure 1.

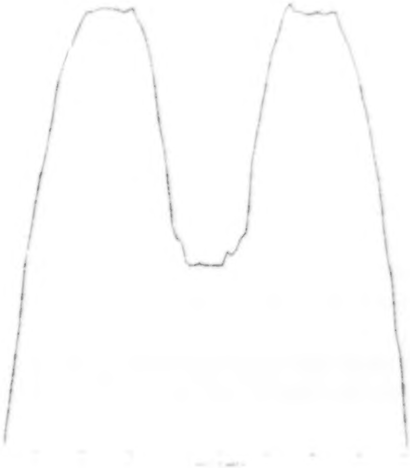

Figure 1.--Fission Fragment Mass Distribution

D. Energy Released in Fission

The most direct method of calculating the energy released in the fission process is to determine the difference in masses of the original nucleus and the resulting products. Although the uranium-235 nuclei may split in many different ways, the products which are obtained in greatest yield have atomic mass numbers of 95 and 139. The initial products of these mass numbers, after several beta emissions, become stable molybdenum-95 and lanthanum-139. A fission which results in these two products would liberate two neutrons.

The mass deficit can be determined by comparing the mass of the reactants with the mass of the products --

Reactants	Mass (amu)
^{235}U	235.044
n	1.009
	236.053

Products	Mass (amu)
^{139}La	138.906
^{95}Mo	94.906
2n	2.018
	235.830

Total Reactants	236.053
Total Products	-235.830
	0.223

Thus 0.223 amu has been converted to energy.

Multiplying 931 MeV per amu times the mass lost (0.223 amu) shows that a total of 208 MeV was released in this reaction. (Usually about 200 MeV is released in uranium-235 fission.) This energy is released as kinetic energy of the fission products, gamma energy, and energy produced by beta decay of the products. Table 1 lists the approximate distribution of fission energy per fission event.

Table 1.--Energy Distribution per Fission Event

Component	MeV
Kinetic energy of fission products . . .	168
Energy of fission neutrons	5
Instantaneous gamma emission	5
Beta particles from fission products. .	7
Gamma rays.	6
Neutrinos from fission products.	10
Total	201

The immediately useful energy is about 174 MeV and eventually a total of about 181 MeV may be available as usable energy. The gamma and neutrino energies usually are not usable.

The gamma and neutrino energy is not readily recoverable. Calculations show that the energy available from the fission of 1 gram of uranium-235 is about 1 megawatt-day.

E. Neutrons Released by Fissioning

Over 99 percent of the neutrons produced in fission are released within 10^{-12} seconds of the fission; these are referred to as prompt neutrons. Less than 1 percent of the neutrons accompanying fission are delayed neutrons, i.e., emitted seconds to several minutes after the fission event. It is the existence of delayed neutrons which allows control of nuclear fission.

F. Maintaining a Nuclear Chain Reaction

To maintain a nuclear reaction one or more neutrons from each fission must initiate another fission. Although an average of 2.43 neutrons per fission are released, the nuclear reaction may not be sustained. Neutrons may leak from the system, be captured by nuclei in (n,γ) reactions, or cause fission. The number of neutrons per fission that cause another fission is called the effective multiplication factor, $k_{(eff)}$. When this factor is equal to or greater than unity, a chain reaction is maintained--that is, a self-sustaining reaction occurs.

G. Production or Breeding

Although uranium-238 undergoes fission when bombarded with high energy neutrons, it is not fissile in the strict meaning. Fissile materials are those which can be split by thermal or slow neurrons. Uranium-238 and thorium-232 are fertile in that, when bombarded by neutrons, they absorb the neutrons, becoming uranium-239 and thorium-233 respectively. Both of these, in a two-step beta decay, produce fissile materials (plutonium-239 and uranium-233).

III. Fusion

A fusion reaction occurs when two light nuclei are forced together to form a heavier nucleus. This newly-formed nucleus is lighter than the sum of the two original nuclei because energy has been released.

Fusion or thermonuclear reactions will be an important source of power in the future. At best, fossil fuels can satisfy the world's demand for power 100 more years. Fissile and fertile fuels can add approximately another 400 years to that; but the supply of deuterium, fuel for fusion, would be sufficient to last the world a billion years. Not only is the supply of deuterium virtually unlimited, but the fusion of deuterium gives off more energy per gram than does the fission of uranium because the fuel is lightweight. Thus, the energy released by the fission of one gram of uranium is equivalent to the energy of 20 tons of TNT; the energy released by fusion of one gram of deuterium is equivalent to the energy released by 57 tons of TNT. As an additional advantage, fusion does not produce intensely radioactive products which must be carefully stored away.

The fusion process, however, is extremely difficult to control. It has been used to increase the yield of "atomic" bombs in creating a thermonuclear weapon, but extensive efforts, over many years, will be required to produce a practical controlled fusion reactor.

A. Controlled Fusion Reactions

There are many problems encountered in initiating controlled fusion reactions. To design a fusion power plant, hydrogen at temperatures ranging from 10 million to 100 million degrees Kelvin must be contained long enough for the fusion of hydrogen nuclei to occur. Some of the possible reactions are as follows:

$$^{2}H + {}^{2}H \rightarrow {}^{3}He + n + 3.267 \text{ MeV energy}$$
$$^{2}H + {}^{2}H \rightarrow {}^{3}H + {}^{1}H + 4.032 \text{ MeV energy}$$
$$^{3}H + {}^{2}H \rightarrow {}^{4}He + n + 17.577 \text{ MeV energy}$$
$$^{3}H + {}^{3}H \rightarrow {}^{4}He + 2n + 11.32 \text{ MeV energy}$$

At such high temperatures all the light atoms are stripped of their electrons to form a gas containing electrons and nuclei. This is called a plasma.

B. Thermonuclear Reactions

The difficulties associated with making a thermonuclear reactor are:

1. maintaining the plasma temperature
2. keeping the plasma pure
3. confining the plasma
4. extracting the energy produced by the fusion.

The first three difficulties have not been effectively solved. The third problem, confinement, is the most difficult. The deuterons must be kept together long enough so that an appreciable number will collide and fuse. This is necessary because the average deuteron (in a plasma containing about 10^{17} nuclei/cm^3 at 10^8 degrees Kelvin) travels 6200 miles before a collision results in fusion. Magnetic fields are being used to try to contain the plasma, thus taking advantage of the fact that all charged particles are influenced by magnetic forces.

The plasma and its confining magnetic field (often referred to as a "magnetic bottle") must be included in a vacuum system. This is to minimize the leakage into the plasma of neutral atoms which come through the magnetic barrier. These impurities cool the plasma by collisions between hot ions and the contaminants and cause charge transfer by which a deuteron or triton collects an electron from the impurity, loses its charge, and escapes.

Heating the plasma to ten million degrees while it is inside the magnetic bottle and in the vacuum chamber was a major problem. There are now many promising ways for solving this problem. Some of these are: accelerator injection, plasma gun injection, and magnetic compression.

The energy released by fusion reaction will eventually be recovered in the form of heat. In half of the deuteron-deuteron reactions and all of the deuteron-triton reactions, most of the energy is carried off by the product neutron. This energy can only be recovered by slowing down the neutron in a blanket of material surrounding the reactor. The energy carried by charged particles can also be recovered as heat, although it may be possible to extract it directly as electricity.

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The Gamma Scintillation Spectrometer

I. INTRODUCTION

Modern electronic circuitry has led to the development of a sophisticated analytical instrument--the gamma scintillation spectrometer. This amazing system cannot only identify gamma emitting radionuclides but can also indicate the activities of each radionuclide.

Samples require very little radiochemical preparation before being counted by the system. Large, bulky samples can be accommodated. Whole-body counters are gamma scintillation spectrometer systems designed to assay the gamma emitting radionuclides in people.

Due to the unusual versatility of this instrument and its recent reduction in cost and size, it is finding widespread application in radiological health work. Although the system appears ideal as an analytical tool, its calibrations and data interpretation are so complex that they comprise a complete field of study referred to as "gamma spectroscopy."

II. COMPONENTS OF GAMMA SPECTROMETER

The gamma scintillation spectrometer operates on the principle that when gamma or X-rays interact within certain crystals, "twinkles" of light or scintillations are produced. The scintillations can be converted into electrical pulses and analyzed. The resulting information can identify the gamma emitting radionuclides and indicate the amount of activity of each.

The gamma spectrometer system is comprised of many parts grouped into three major functional systems: the detector system, the pulse analyzer system and the display system. (See Figure 1.)

III. THE DETECTOR SYSTEM

A. The Source

An investigator has wide latitude in choosing his source of radiation. Any source emitting X-ray or gamma radiation can be used, since self-absorption of gamma photons within a gamma source is relatively low and can be compensated for in calibration. Large, bulky samples can be used with relatively little preparation.

B. The Shield

Naturally-occurring gamma emitting nuclides in building materials, the earth, and

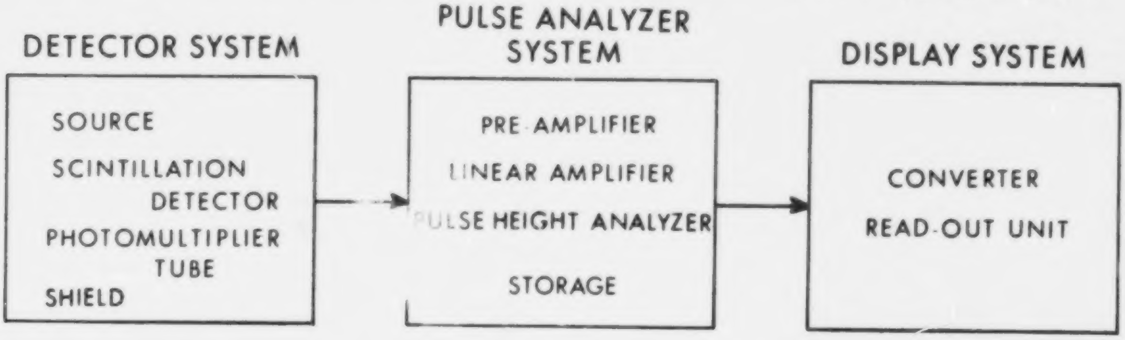
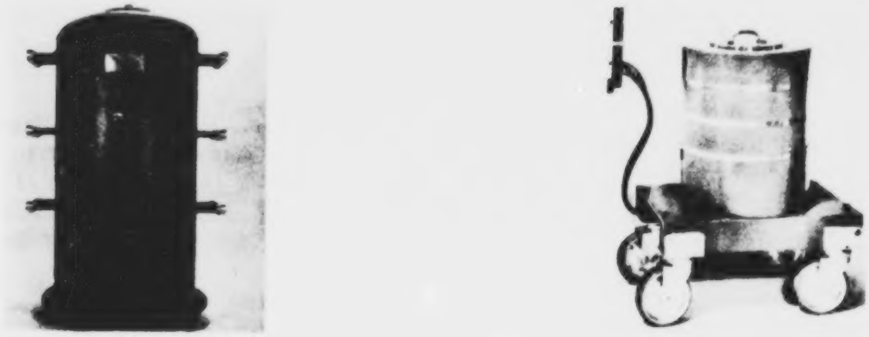


Figure 1.--Gamma Spectrometer Components

(a) 5-Ton Steel Shield External View (Courtesy of Instrument & Development Products Co.)

(b) 5-Ton Steel Shield Internal View (Courtesy of Instrument & Development Products Co.)



(c) 1000-Pound Lead Shield Module Construction (Courtesy of Instrument & Development Products Co.)

(d) 1000-Pound Lead Shield Mobile (Courtesy Victoreen Instrument Co.)

(3) 4-Ton Steel Shield

(f) 6-Ton Steel Shield

Figure 2.--Gamma Detector Shields

other surroundings may necessitate a rather massive shield to reduce detector interactions from origins other than the intended source. The shields must be massive due to the exceptional penetrating power of gamma photons. Such shields are only used where exceptional sensitivity is required. They are usually constructed of steel or lead and lined with copper or steel. Figure 2 shows some shield designs.

C. The Crystal

The sensing unit of the spectrometer is the crystal. It is especially chosen for its ability to produce scintillations when photons interact within its structure. The most common choice of crystals for gamma ray detection is the sodium iodide, thallium activated, [NaI(Tl)] crystal. This is a carefully "grown" crystal produced with minute traces of a specific impurity known as an "activator." The activator enhances the crystal's ability to scintillate visible light.

Crystals are encased in light-tight cans, usually of aluminum or stainless steel. This assures that the only light viewed by the photomultiplier is that which is produced by radiation interacting within the crystal. (See Figure 3.)

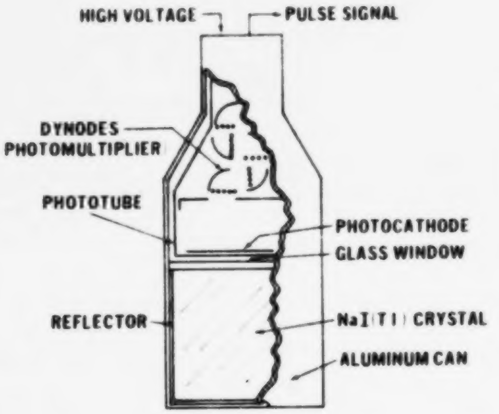


Figure 3.--Detector Crystal and Photomultiplier Assembly

In the case of NaI(Tl) crystals it is important that the crystal be hermetically sealed to keep out moisture. The sodium iodide is hygroscopic and collects moisture if exposed to the atmosphere. If water droplets form on the crystal, they will gradually dissolve it.

D. Interaction in the Crystal

With detectors constructed as shown in Figure 3, the canning will keep charged particulate radiation from reaching the crystal. Thus, interactions within the crystal are solely due to gamma or X-radiation.

Photons, on entering the crystal, can either interact or pass through undisturbed. Although most photons do pass through, a sufficient number of photons interact with the crystal atoms. It is important to recall that gamma and X-ray photon interactions with matter occur primarily by three modes: photoelectric effect, Compton scattering, and pair production. In each of these interactions, photon energy is imparted to either an electron or a positron. It is the movement of these charged particles through the crystal lattice, exciting the atoms and molecules, that initiates the scintillation process. The crystal atoms receive energy from a speeding charged particle passing nearby. The energy transfer slows the passing particle slightly. The atom absorbs the energy by slightly increasing the energy state of one or more of its more loosely bound electrons. Since these electrons cannot exist very long in this excited state they will quickly release the excess energy by emitting a light photon and return to a more stable energy state. The entire light output of one flash is really the total of millions of such events occurring within 10^{-12} seconds.

The three interaction processes cause sequential events to occur differently. But they frequently accomplish a similar end--total transfer of photon energy to the crystal and hence the occurrence of scintillations.

1. The photoelectric process is one in which the total photon energy is transferred directly to an orbital electron. Although the

photoelectron may cause secondary ionization, all the freed electrons will be slowed down and will give up their energy through excitation processes to produce light. These events happen so quickly that they appear as a single flash of light representative of the entire energy of the photon.

2. The Compton scattering process results in only a portion of the photon energy being imparted to the orbital electron. The scattered photon with reduced energy may pass out of the crystal or it may interact elsewhere in the crystal and undergo a photoelectric or another Compton scattering process. If both the initial and the scattered photons interact in the crystal, the processes occur so quickly that again it appears as one event of total energy transfer.

3. The pair production process results in the conversion of 1.02 MeV of the photon energy to the mass of an electron and positron. The remainder of the photon energy is divided between the two particles as kinetic energy. The positron, upon slowing down, interacts with an electron and this annihilates both of them. In their place, two photons of 0.51 MeV each are produced. Either one or both of these photons may interact in the crystal by processes (1) or (2). Again, the total chain of events appears as one event. In one case, the scintillation might be indicative of the total initial photon energy ($E\gamma$). Since there is also a probability that one or both of the photons from the positron annihilation might be lost, the light intensity may be indicative--($E\gamma$ = .51 MeV, one escaping photon) or ($E\gamma$ = 1.02 MeV, two escaping photons). The most probable of these three occurrences will be a total energy transfer event ($E\gamma$).

E. The Photomultiplier Tube

The light energy produced in the crystal is converted into electrical energy by a photomultiplier tube, composed of a photocathode and an electron multiplier. When struck by light, the photocathode emits electrons. The number of electrons released is proportional to the amount of light incident upon the photocathode. The small number of electrons released is then multiplied by accelerating them across a series of voltage potentials.

An external unit, the high voltage power supply, furnishes a large voltage potential to the photomultiplier tube. This power supply can be either a battery pack or an electronic instrument which delivers a pre-selected, very stable voltage between 500 and 2000 volts. Small variations in the applied voltage result in large errors in the accumulated data.

Summarizing, then: (1) a gamma interaction has been converted into a flash of light; (2) the light is converted into freed electrons; (3) the electrons are multiplied; and (4) the resultant group of electrons comprise an electrical pulse whose size or voltage is proportional to the gamma photon energy received by the crystal.

IV. THE PULSE ANALYZER SYSTEM

A. Preamplifier

Although we have an electrical pulse at the output of the photomultiplier tube, it is still weak and could be easily lost if transmitted over any significant length of cable. In order to maintain a reliable flow of information, a preamplifier is installed close to the photomultiplier tube. This shapes and strengthens the signal for the trip to the remainder of the system.

B. Linear Amplifier

The initial stage in the analyzer instrument cabinet is a linear amplifier. It receives the pulse from the preamplifier by cable leads; it must amplify or enlarge each pulse by the same factor. A precision control is available to adjust the amplification.

C. The Pulse Height Analyzer

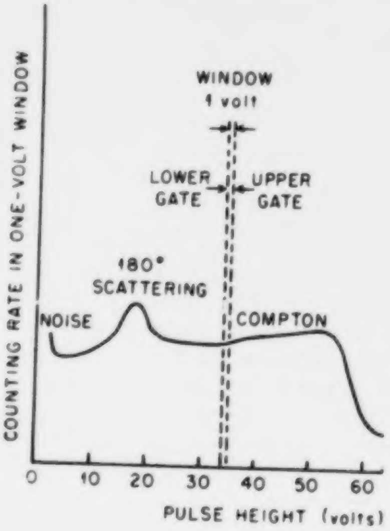
After the pulse leaves the linear amplifier, it enters the pulse height analyzer to be sorted according to size and, therefore, according to the energy it represents. Analyzers are classified as either voltage discriminator types or computer types.

1. Voltage Discriminator

The voltage discriminator analyzer measures the pulse by a voltage "mouse trap" technique. An electronic "trap" is designed for each pulse sorting category or channel. This is done by electronically requiring a pulse to be of a minimum size or voltage to open the "gate" and enter the trap. An additional gate is supplied such that if the pulse is sufficiently large it can open this second "gate" and let itself out. When a pulse is "trapped," a count is recorded in the instrument for that "mouse trap" or channel.

Some instruments may contain several channels in succession so that a particular gate serves as the "out" gate for the preceding channel and the "in" gate for the next channel. The difference in voltage requirements (the "out" gate minus the "in" gate) is referred to as the "window width." Figure 4 demonstrates this on a typical plot of data.

The lowest gate in the instrument is called the low-level discriminator. The uppermost gate is called the high-level discriminator.



Analyzers can contain one or more channels; they are called single or multichannel analyzers. The recording device for the voltage discriminator analyzer is a scaler or series of scalers. Figure 5 shows two

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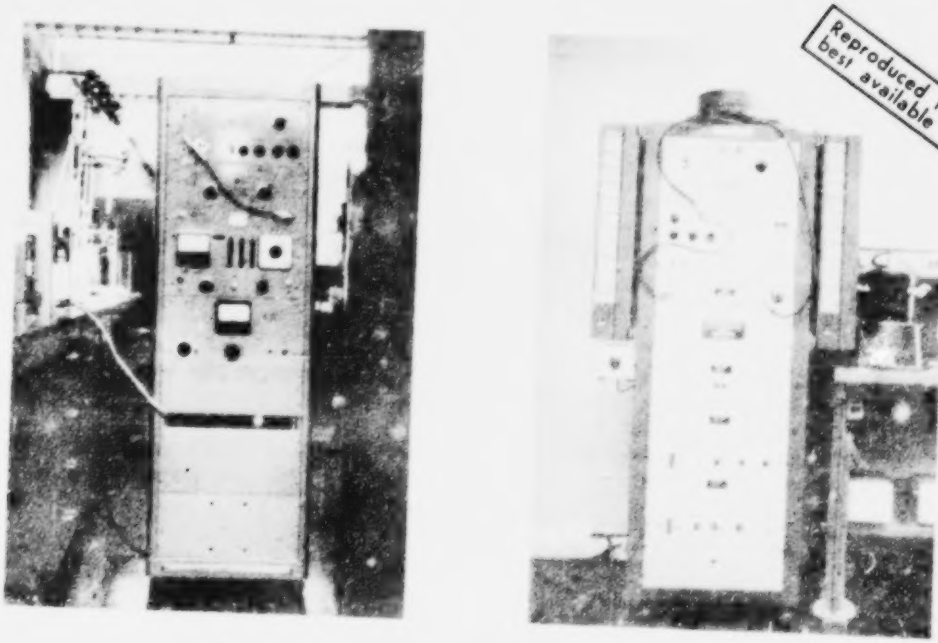

Figure 5.--Voltage Discriminator Analyzers

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examples of voltage discriminator analyzers. This type of circuitry is limited primarily to single-channel analyzers in the current models. Some of the older multichannel analyzers used the "mouse trap" circuit.

2. Computer-type Analyzer

The advent of modern digital computers has significantly advanced analyzer designs. Nearly all multichannel analyzers use computer circuitry. These analyzers achieve voltage discrimination by first converting the pulse size into a coded number. This is accomplished by an oscillator which times the rate of rise of the pulse. If the oscillator completes 50 cycles during the pulse rise, a signal adds one count in channel #50. Likewise, 25 oscillations indicate a count for channel #25. The oscillator will run through only as many cycles as the analyzer has channels. If the pulse is still not classified at the end of the oscillator run, the count is rejected.

The pulse size information now has been converted to a digital number. This is called "analog to digital conversion." The digital number signals the computer circuit to store an additional tally in the memory for the channel of the same number.

The memory circuits are like those of other computers. Here they serve as a multitude of scalers, one for each channel. Such circuits are easily compatible with other, more sophisticated computers so data can be readily fed from one to another.

Figure 6 shows an older vacuum tube model of this type analyzer. With solid state circuitry, increased reliability and dependability have become available in compact units of lower costs. Some examples of these are shown in Figures 7, 8, 9, and 10.

Figure 6.--Vacuum Tube Computer Type Analyzer

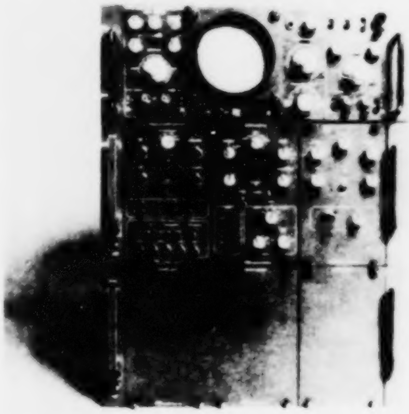

Figure 7.--Pulse Height Analyzer (Courtesy of Nuclear Data, Inc.)

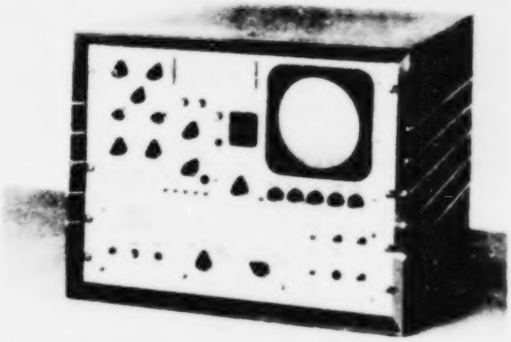

Figure 8.--Pulse Height Analyzer (Courtesy of Victoreen Instrument Co.)

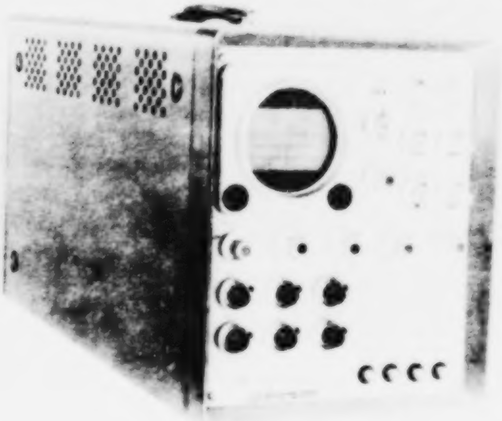

Figure 9.--Pulse Height Analyzer (Courtesy of Technical Measurement Corporation)

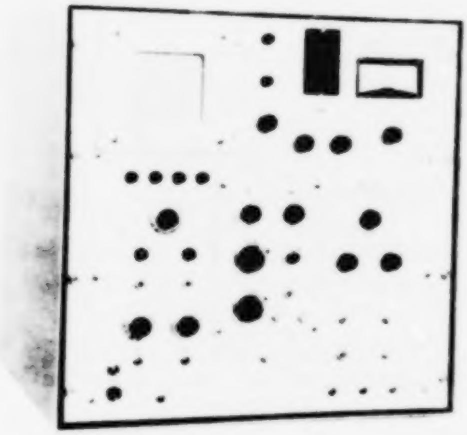

Figure 10.--Pulse Height Analyzer (Courtesy of Radiation Instrument Development Laboratory)

V. DISPLAY SYSTEM

The accumulation of data in an analyzer memory section is useless without some provision for presenting or displaying such information. This is accomplished by a "read-out" system. Usually several read-out systems are used with an analyzer. The choice of a system is suggested by the manner in which the information is to be used. Some examples of these are shown in Figures 11 through 19.

Examples of Read-Out Devices

Device	Mode of Read-Out	Figure
Oscilloscope	Plotted light dots on a cathode ray tube.	11
Strip chart recorder	A line graph inscribed on a moving strip of graph paper.	12
X - Y plotter	Automatically plotted points on graph paper.	13
Scaler	Lights and register indicate counts in a selected channel.	--
High-speed printer	Prints out channel number and channel counts on a strip of paper.	14
Typewriter	Types channel number and channel count on paper.	16
Punch tape*	Coded holes punched in paper tape to represent channel number and channel count.	17 18
Magnetic tape*	Coded magnetic bits are arranged on magnetic recording tape to represent channel number and channel count.	19

*These read-outs facilitate the use of data on larger, more "general purpose" computers.

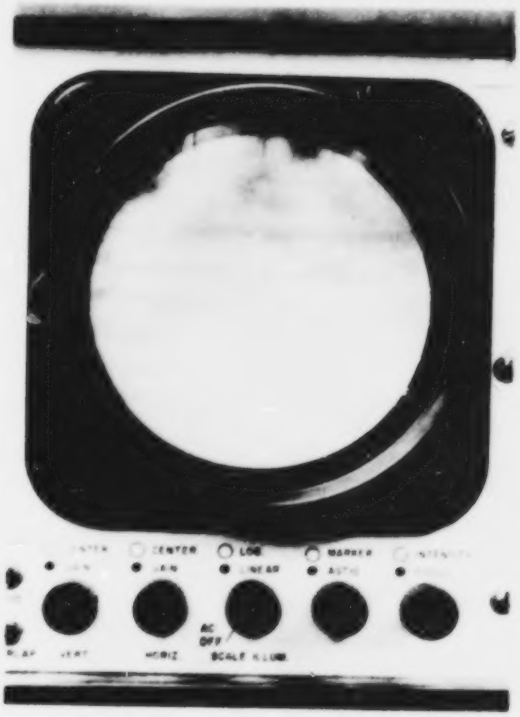
CENTER
CENTER
LOG
MARKER
INTENSITY
GAIN
GAIN
LINEAR
AUTO
AC
OFF
VERT
HORIZ
SCALE ILLUM

Figure 11.--Oscilloscope.

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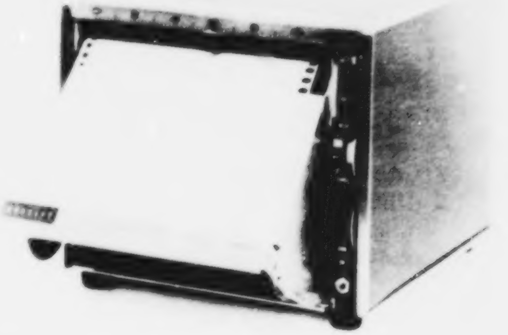

Figure 12.--Strip Chart Recorder (Courtesy of Hewlett-Packard/Moseley Div.)

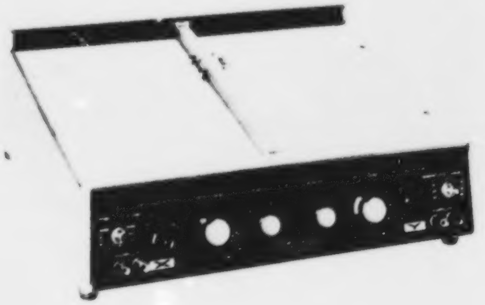

Figure 13.--X-Y Plotter (Courtesy of Hewlett-Packard/Moseley Div.)

Figure 14.--Parallel Printer (Courtesy of Hewlett-Packard Company)

```
006        00549
005        00691
004        00474
003        00106
002        00067
001        00000
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Figure 15.--Parallel Printer Tape Format

Figure 16.--Electric Typewriter (Courtesy of International Business Machines)

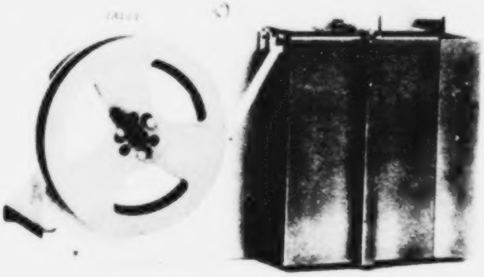

Figure 17.--Paper Tape Perforator (Courtesy of the Tally Corporation)

Figure 18.--Paper Tape Reader (Courtesy of the Tally Corporation)

Figure 19.--Magnetic Tape Unit (Illustration reproduced with permission from the General Electric Computer Dept.)

VI. SUMMARY

The gamma spectrometer is a sophisticated analytical instrument used to identify and assay gamma-emitting radionuclides. This is accomplished by electronically analyzing light scintillations resulting from gamma- and X-ray... interactions within specially designed crystals. The use and interpretation of gamma spectrometer data is referred to as gamma spectroscopy.

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This material was prepared by R. L. Shearin formerly of the Training Branch of the Division of Radiological Health.

Assay of Radionuclides: Gamma Emitters

I. INTRODUCTION

Because there are definite energy levels in the nucleus, gamma photons from a given radionuclide are emitted with discrete characteristic energies. As a result, determination of gamma energies through use of a gamma scintillation spectrometer can contribute materially to the identification of such nuclides.

The energies of gamma rays from any one nuclide, however, are not unique. Hence, for positive identification of a given radionuclide, gamma spectral data must be augmented by determination of half-life and maximum beta energy, or by chemical separation.(1)

II. CHEMICAL SEPARATION

In cases wherein gamma energies are known but the identity of a nuclide remains uncertain, chemical separation may be used to confirm identification.

In cases wherein a single sample contains so many different gamma-emitting radionuclides that accurate interpretation of spectral data is difficult if not impossible, chemical separation may be required as a prior step to the determination of gamma energies. In initial separations of such samples, a general chemical procedure must often be employed. One separation scheme which has proved effective for certain of the fission products as well as neutron induced radionuclides is shown in Table I(2). The scheme outlines steps for the separation of the gross sample into six chemical groups:

A. Silver Group
B. Acid Sulfide Group
C. Hydroxide Group
D. Basic Sulfide Group
E. Alkaline Earth Group
F. Soluble Group

If, through study of the spectral curves of any one chemical group, the presence of a radioisotope of a given element is suspected, additional specific chemical separation techniques may be employed to confirm or deny its presence.

III. QUANTITATIVE DATA

To determine disintegration rates of radionuclides from spectral data, the gamma spectrometer must be calibrated with similar sources of known intensities. The unknown and known sources are then related through comparison of photopeak areas. Such measurement may be done by cutting out and weighing the peak areas, or by use of a planimeter.(3)

A secondary method which may be used to estimate quantitative data for the several nuclides in a given sample is to follow their decay on a gross gamma counter. The relative amounts of the several nuclides in the sample can be determined from the plotted data. Through suitable application of chemical yield and counting efficiency factors, a quantitative estimate of each radionuclide in the sample is then made possible.

IV. INTERPRETATION OF SPECTRAL DATA

On the following pages are presented spectral and gross gamma decay data for a

Table I

CHEMICAL SEPARATION SCHEME

1) ADD AN ALIQUOT OF THE RADIOACTIVE SAMPLE TO A 50 MILLILITER CENTRIFUGE TUBE. AERATE TO REMOVE RADIOACTIVE GASES. ACIDIFY WITH HNO_3 AND ADD Cl, Br, I, Cu, As, Fe, Y, Co, Mn, Zn, Ba, Sr, Ca, Na, AND K CARRIERS. PRECIPITATE Cl, Br, AND I WITH SLIGHT EXCESS OF Ag. HEAT TO COAGULATE THE PRECIPITATE. CENTRIFUGE.

2) PRECIPITATE: AgCl, AgBr, AgI, WASH WITH WATER, FILTER, MOUNT AND COUNT. DESIGNATE AS THE SILVER GROUP.

3) SUPERNATE: ADJUST ACIDITY TO 0.5 N HCl. SATURATE WITH H_2S. CENTRIFUGE.

4) PRECIPITATE: $CuS + As_2S_3$. WASH WITH WATER, FILTER, MOUNT AND COUNT. DESIGNATE AS THE ACID SULFIDE GROUP.

5) SUPERNATE: TRANSFER TO BEAKER. ADD 1 ml. CONC. HCl. BOIL UNTIL SOLUTION IS CLEAR. OXIDIZE Fe^{++} TO Fe^{+++} BY BOILING WITH HNO_3. COOL SOLUTION AND MAKE 1:1 WITH CONC. HCl. NEUTRALIZE WITH CONC. NH_4OH TO PRECIPITATE HYDROXIDE. CENTRIFUGE.

6) PRECIPITATE: Fe + Y HYDROXIDES. WASH WITH WATER, FILTER, MOUNT AND COUNT. DESIGNATE AS THE HYDROXIDE GROUP.

7) SUPERNATE: SATURATE WITH H_2S. CENTRIFUGE.

8) PRECIPITATE: CoS, MnS, ZnS. WASH WITH WATER, FILTER, MOUNT AND COUNT. DESIGNATE AS THE BASIC SULFIDE GROUP.

9) SUPERNATE: ACIDIFY SOLUTION WITH HCl. BOIL TO REMOVE H_2S. ADD $(NH_4)_2H(PO_4)$ AND AN EXCESS OF NH_4OH. DIGEST. CENTRIFUGE.

10) PRECIPITATE: Ca, Ba, AND Sr PHOSPHATES. WASH WITH WATER CONTAINING A FEW DROPS OF NH_4OH FILTER, MOUNT AND COUNT. DESIGNATE AS THE ALKALINE EARTH GROUP.

11) SUPERNATE: Na + SOLUBLE IONS. TRANSFER ALIQUOT TO CULTURE TUBE FOR COUNTING. DESIGNATE AS THE SOLUBLE GROUP.

sample of reactor cooling water which contained gamma-emitting fission products as well as neutron induced radionuclides. The sample, 140 milliliters in volume, was separated into six chemical groups following the general scheme given in Table I. Each group was analyzed spectrally on a 20-Channel gamma scintillation spectrometer. Concurrently, the decay of each group was followed on a gross gamma counter.(2)

Gamma spectral and decay data for the Silver Group are shown in Figures 1 and 2. The spectral data shows that 21 h I^{133}, 8.05 d I^{131} and their daughter products constitute a major portion of this chemical group. The presence of I^{133} is characterized by the prominent 0.53 Mev gamma peak and the lesser 0.87 and 1.31 peaks as shown in Curve B of Figure 1. Iodine-131 is characterized by the prominent 0.37 and 0.64 Mev peaks in Curve C of the same figure, as well as the approximately 8 d half-life shown in Figure 2. The 0.08 Mev peak is representative of 5.27 d Xe^{133} and its daughter Cs^{133m} as well as 4.8 x 10^{-10}s Xe^{131m}1, daughter products of I^{133} and I^{131}, respectively.

The prominent 0.25 Mev peak in Curves A and B is characteristic of 9.1 h Xe^{135} + Cs^{135m}, which strongly suggests the presence of its parent 6.7 h I^{135}. Iodine-135 is confirmed by the presence in Curve B of peaks near its characteristic gamma-energies of 0.84, 1.13, 1.28, 1.46, and 1.72 Mev(4).

The rapid initial gross gamma decay of the sample as shown in Figure 2 strongly suggests also the presence of 52 m I^{134}. However, its presence would be difficult to confirm by the spectral data.

The rapidly decaying 0.66 Mev gamma peak in Curve A is indicative of 2.3 h I^{132}. Similarly, the intermediate gross gamma decay shown in Figure 2 also suggests the presence of I^{132}.

Gamma spectral data for the Acid Sulfide Group are shown in Figure 3. The 0.51 and 1.03 Mev peaks indicate the presence of the positron emitter 12.8 h Cu^{64}. This is further substantiated by the presence of the 1.34 Mev gamma peak. The 0.14 Mev peak is indicative of 6.0 h Tc^{99m}. Gross gamma decay data shown in Figure 4 confirm these indications.

Gamma spectral data for the Hydroxide Group are shown in Figure 5. The 0.10 and 0.22 Mev peaks are indicative of 2.33 d Np^{239}. Confirmation is given by the gross gamma decay data as shown in Figure 6 and the subsequent single channel spectrum showing the characteristic 0.09, 0.21 and 0.28 Mev gamma peaks as shown in Figure 7. The 0.53 Mev peak is indicative of 1.87 h F^{18} and 50 m Y^{91m}. The 0.83 Mev peak represents 2.58 h Mn^{56}. The 1.36 Mev peak is possibly indicative of 15.0 h Na^{24}. Yttrium-91m, Mn^{56} and Na^{24} are confirmed in later chemical groups.

Gamma spectral data for the Basic Sulfide Group are shown in Figure 8. The 0.10 and 0.22 Mev peaks again show the presence of 2.33 d Np^{239}. The 0.84 and 1.78 Mev peaks and gross gamma decay data as shown in Figure 9 confirm the presence of 2.58 h Mn^{56}. The long lived 1.16 and 1.34 Mev peaks and the chemistry of this particular sample confirm the presence of 5.2 y Co^{60}.

Gamma spectral data for the Alkaline Earth Group are shown in Figure 10. The 0.10 and 0.22 Mev peaks of Curve A are again indicative of 2.33 d Np^{239}. Half-life studies of the 0.56, 0.76, and 1.05 Mev peaks showed them to be associated with 9.7 h Sr^{91} + Y^{91m}. As was [illegible]tioned earlier, evidence for 50 m Y^{91m} has been found in the Hydroxide Group. The 1.39 Mev peak is possibly indicative of 2.7 h Sr^{92}. The 0.10 and 0.30 Mev peaks shown in Curve C of the same figure are representative of 27.4 d Pa^{233}. The 0.16 Mev peak is indicative of

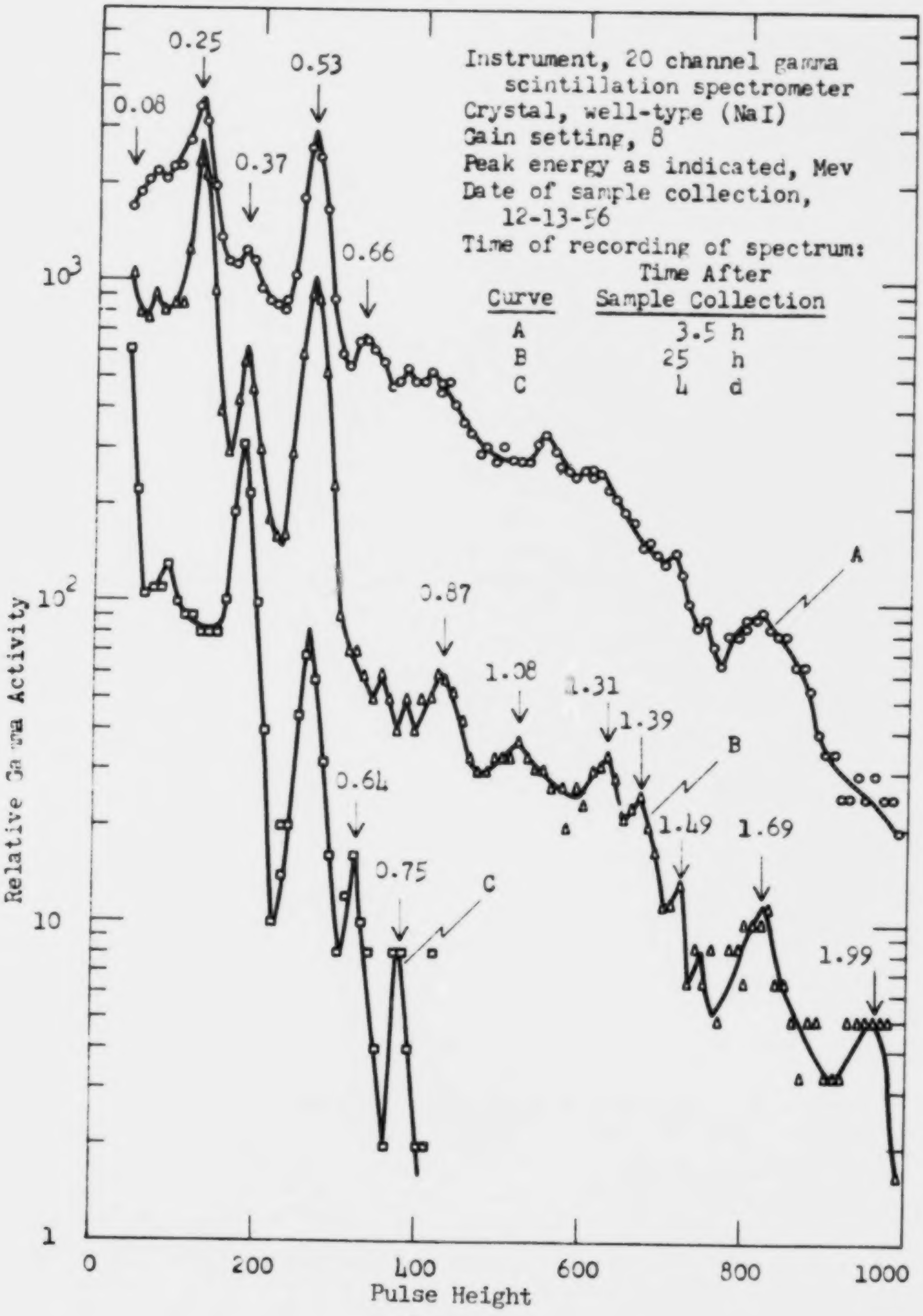

Figure 1

Gamma spectral data for reactor cooling water: Silver group.

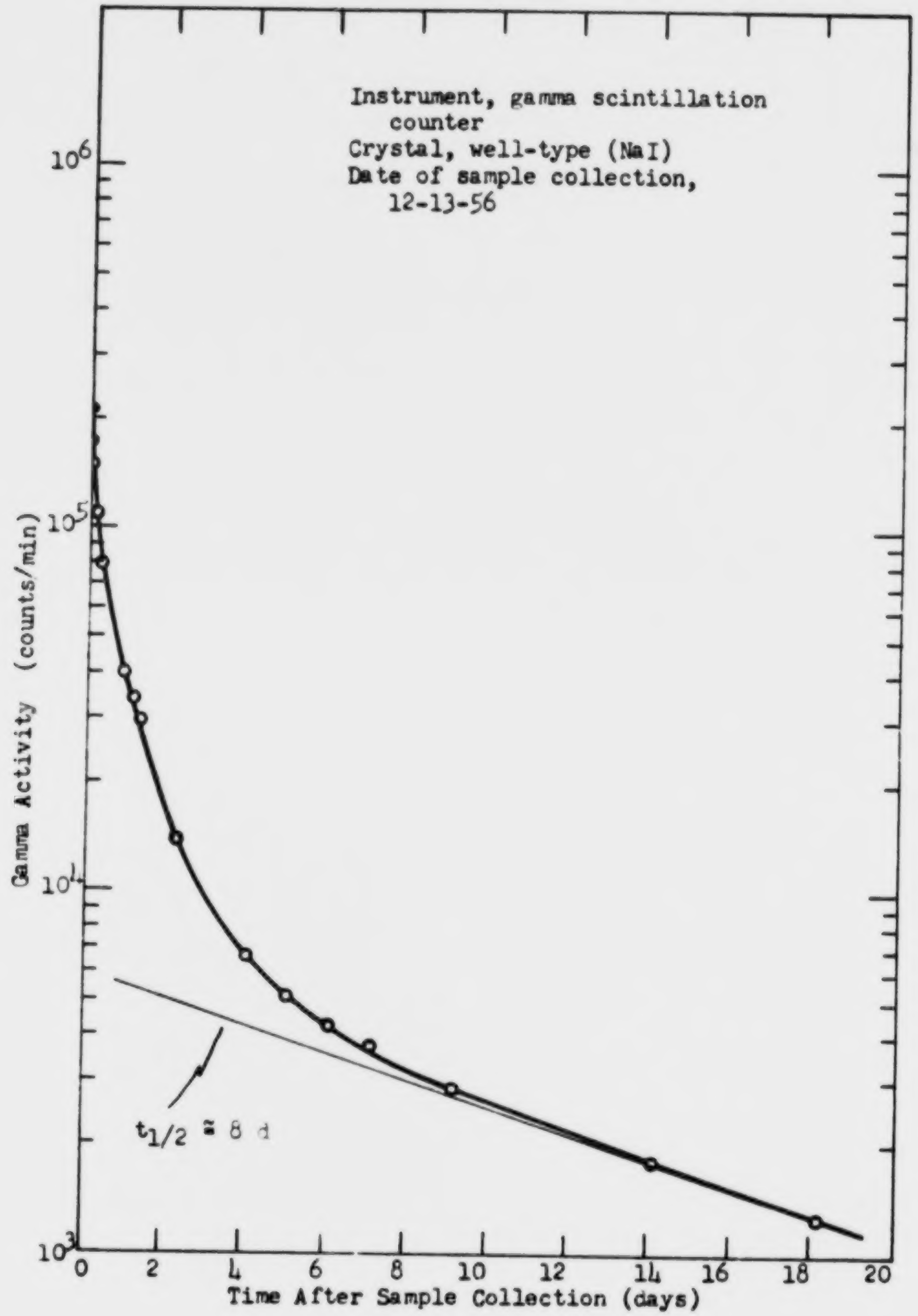

Figure 2

Gross gamma decay for reactor cooling water: Silver Group.

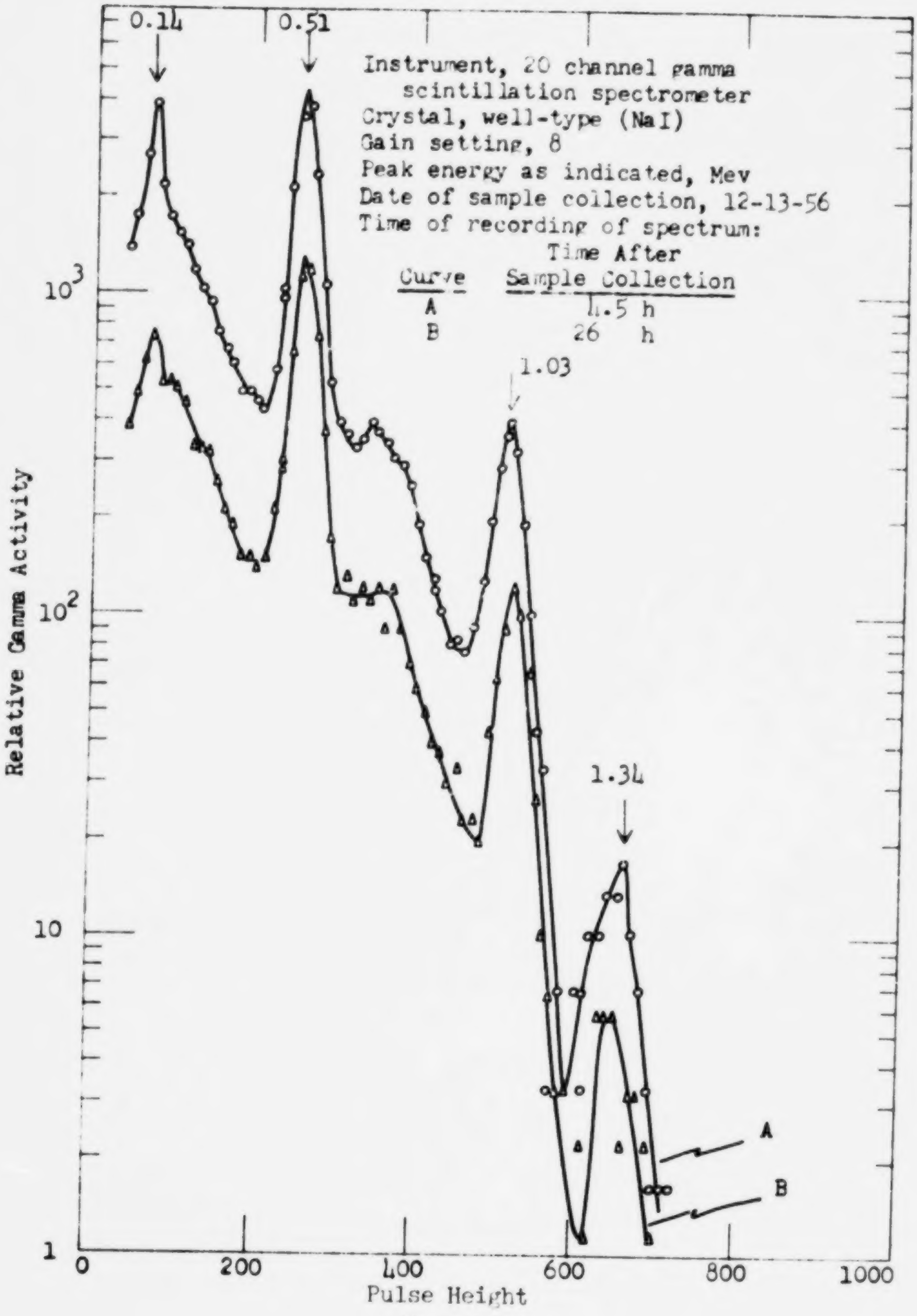

Figure 3

Gamma spectral data for reactor cooling water: Acid sulfide group

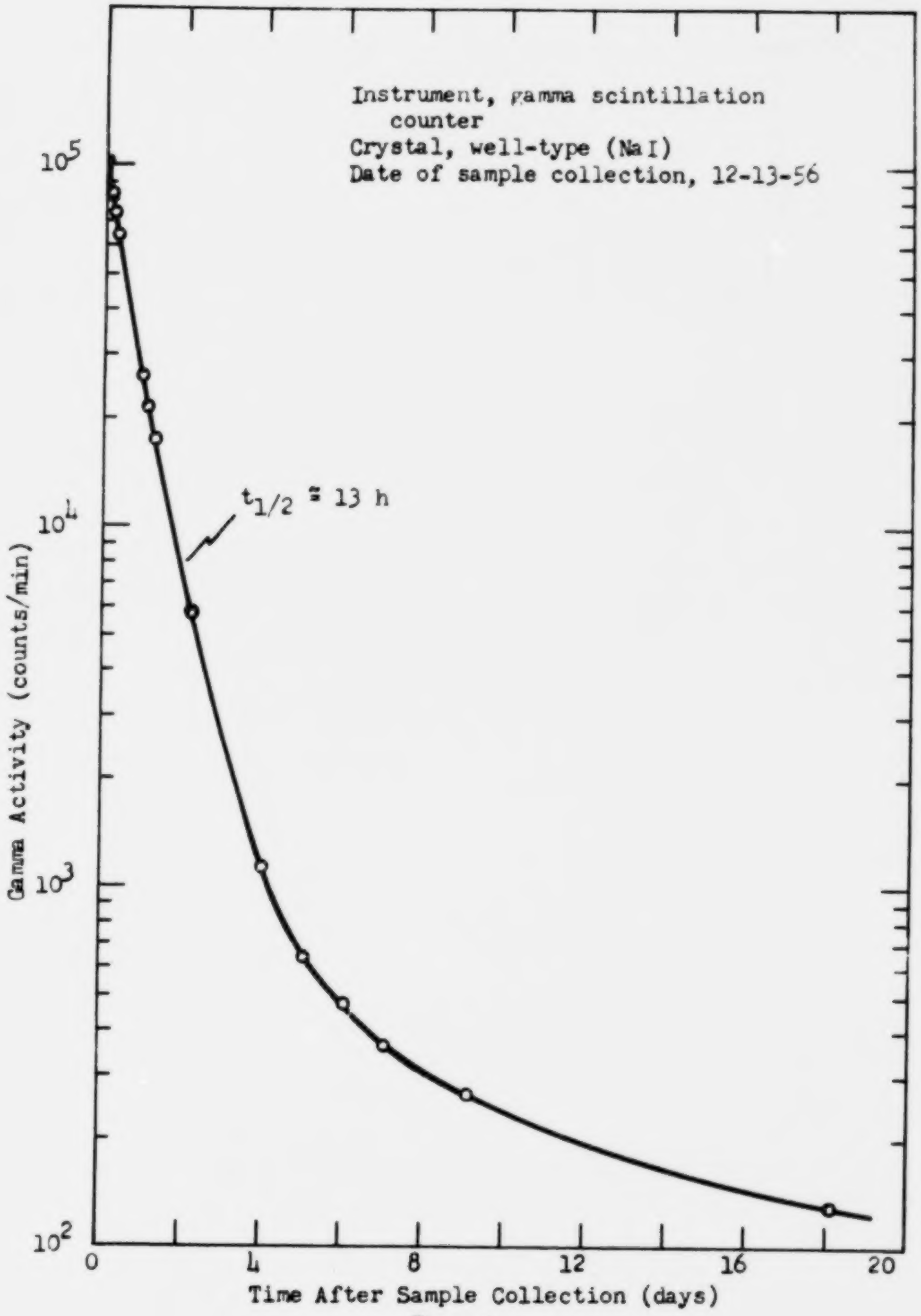

Figure 4

Gross gamma decay for reactor cooling water: Acid sulfide group.

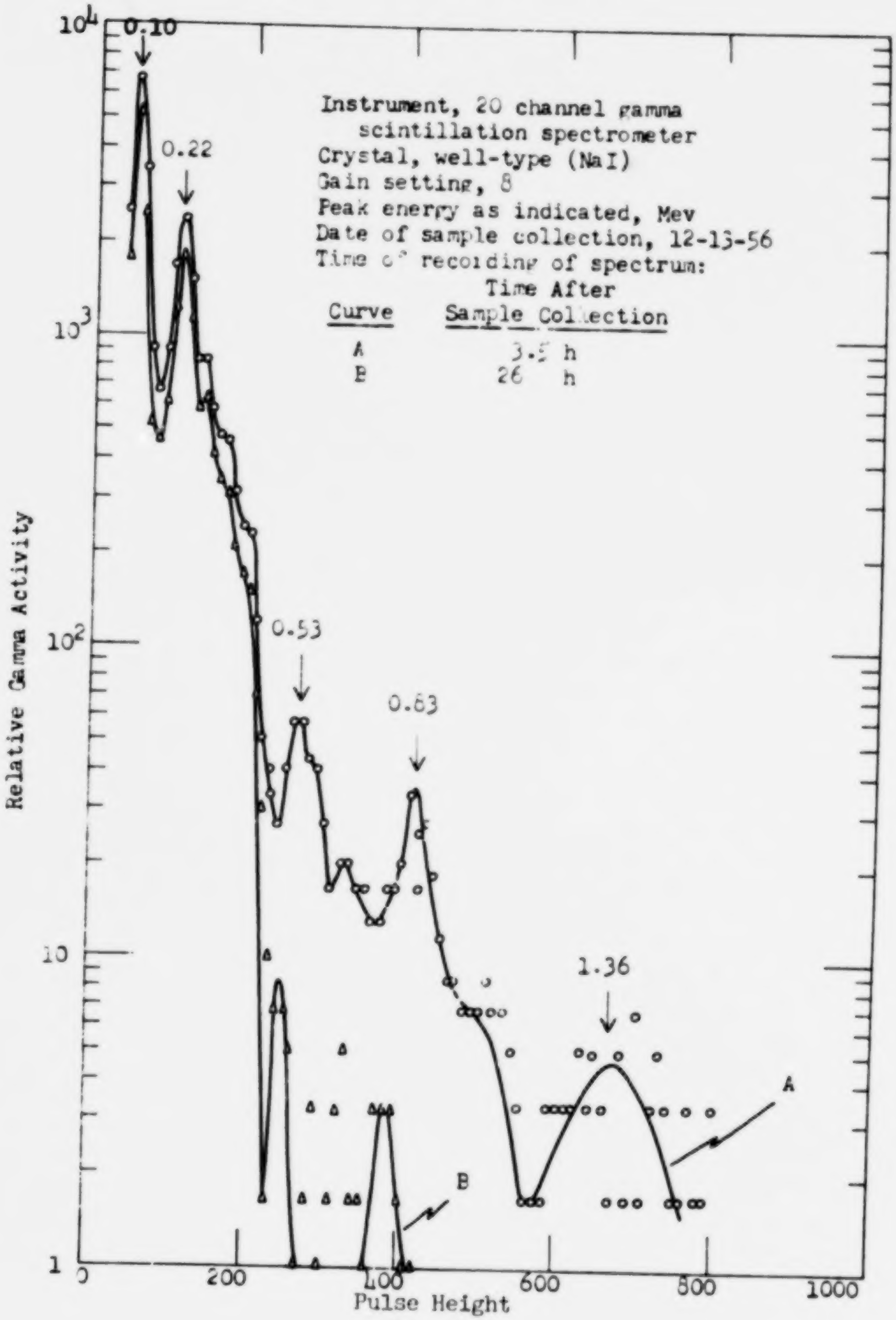

Figure 5

Gamma spectral data for reactor cooling water: Hydroxide group.

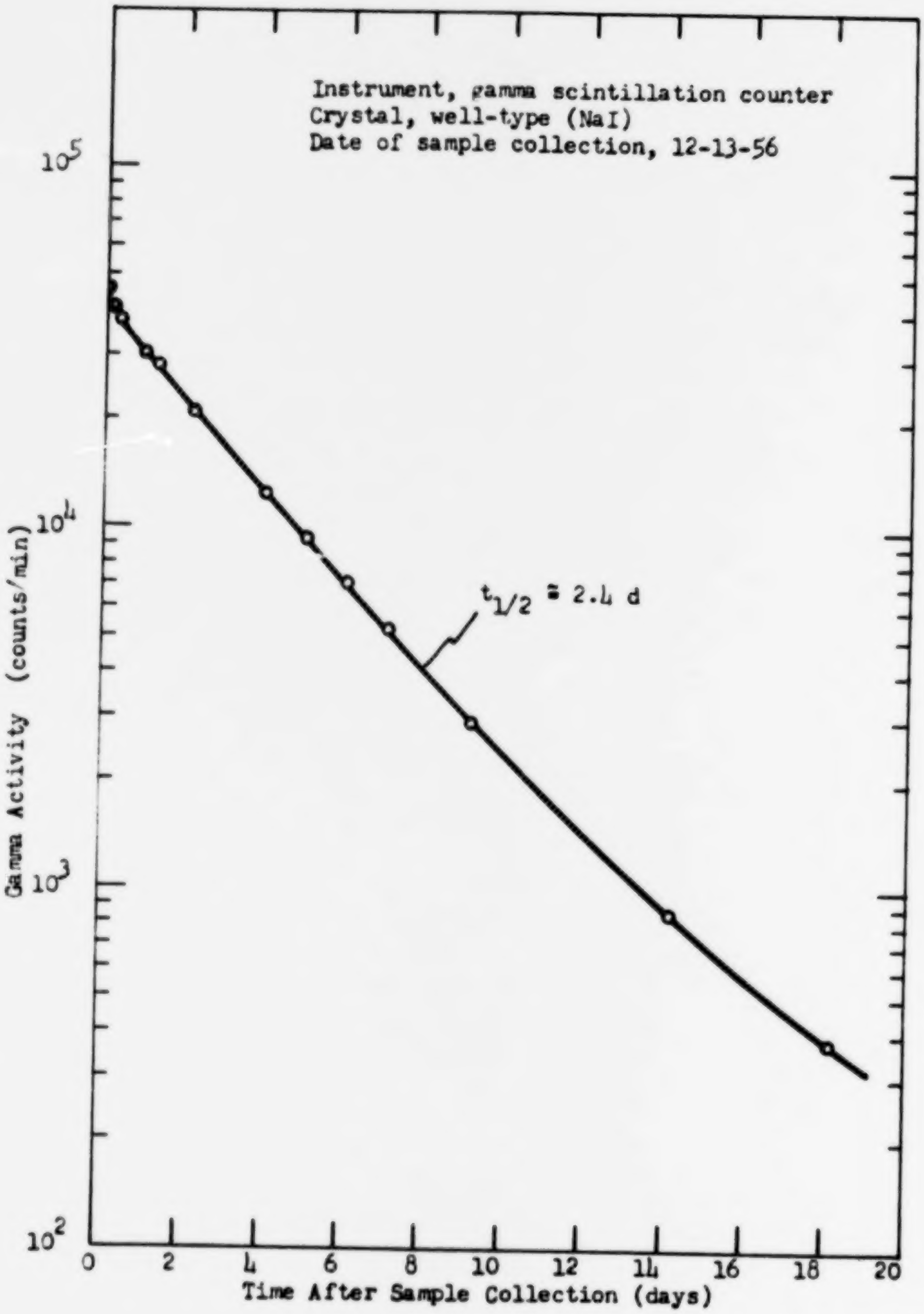

Figure 6

Gross gamma decay for reactor cooling water: Hydroxide group.

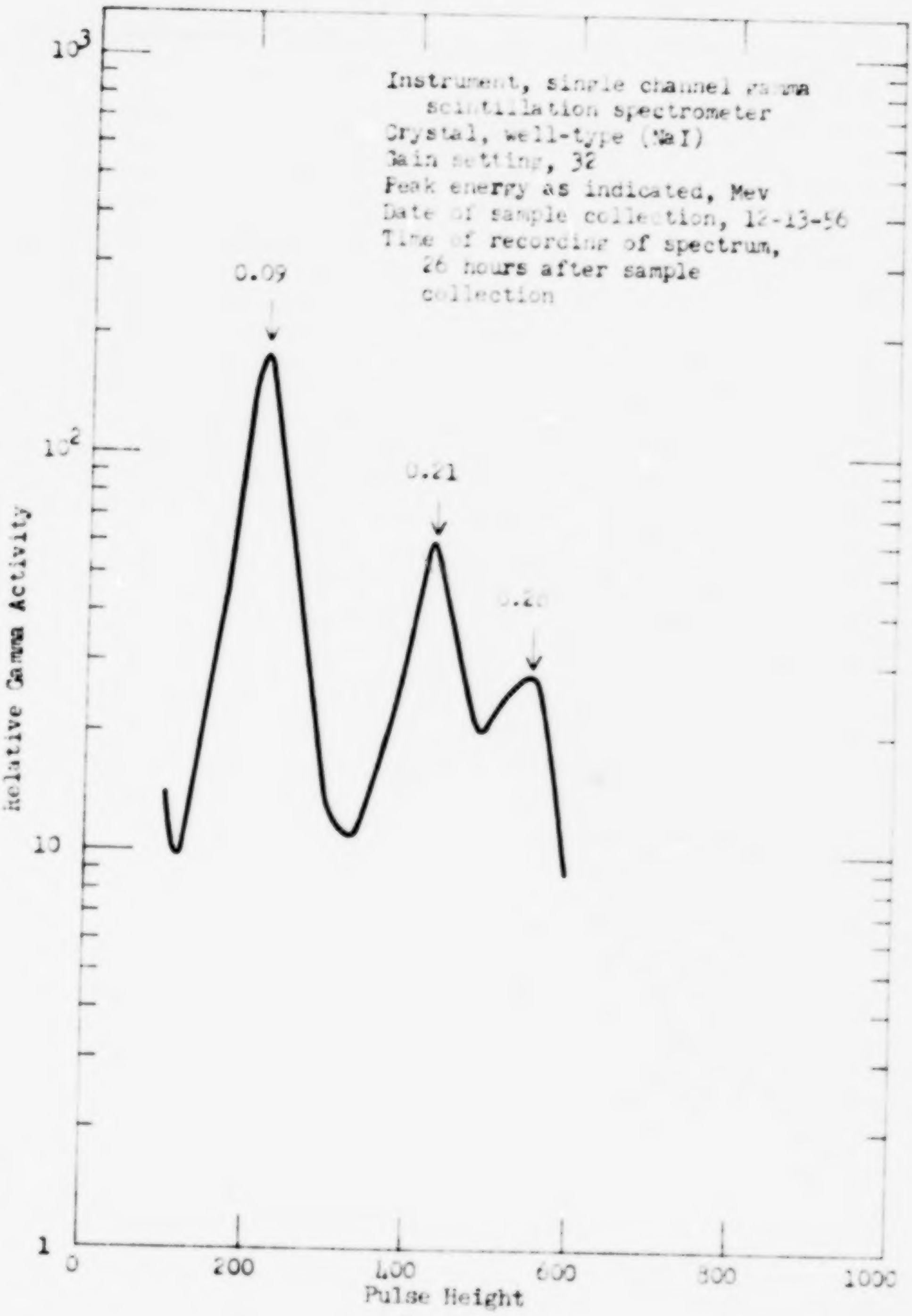

Figure 7

Gamma spectral data for reactor cooling water: Hydroxide group.

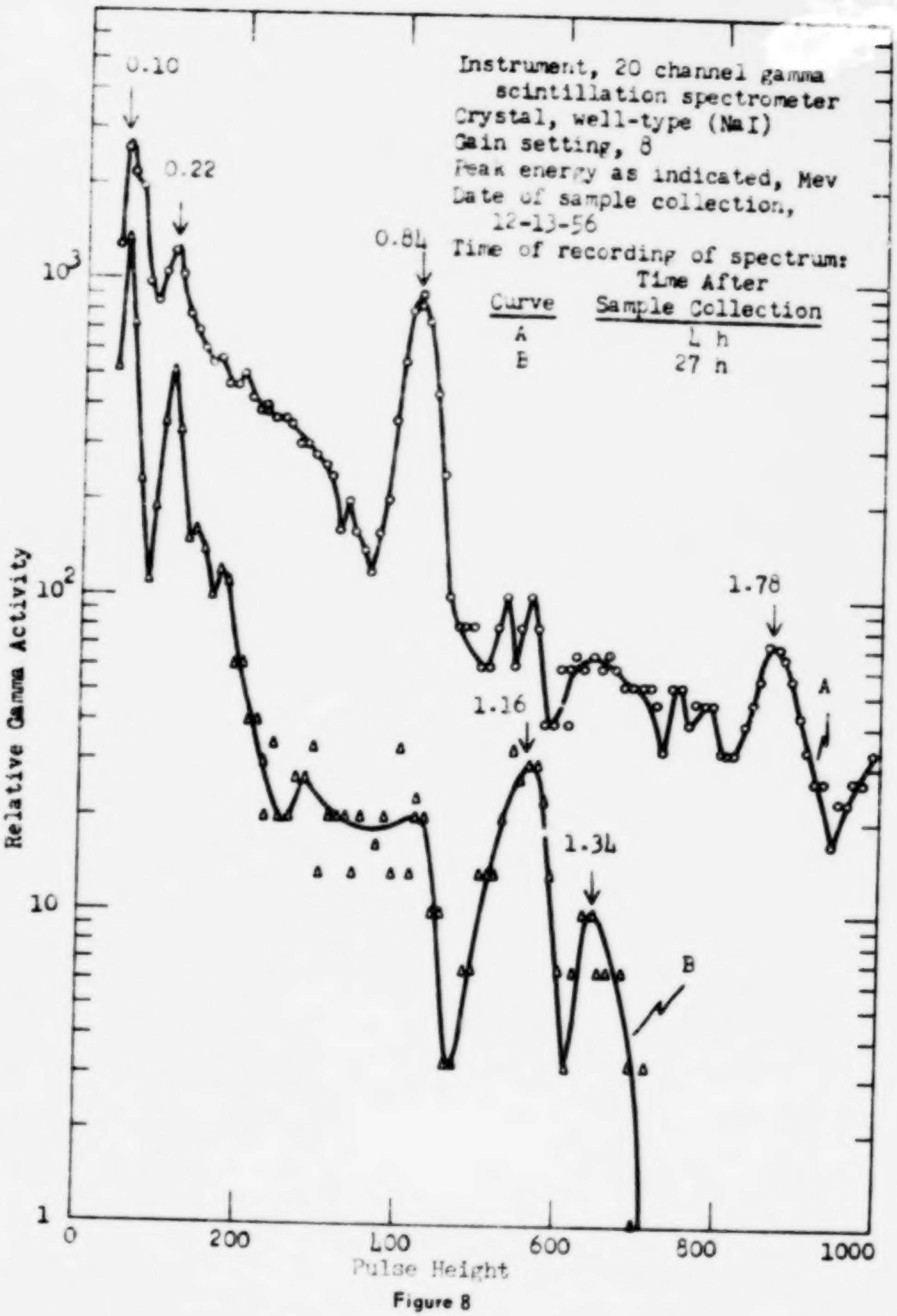

Figure 8

Gamma spectral data for reactor cooling water: Basic sulfide group.

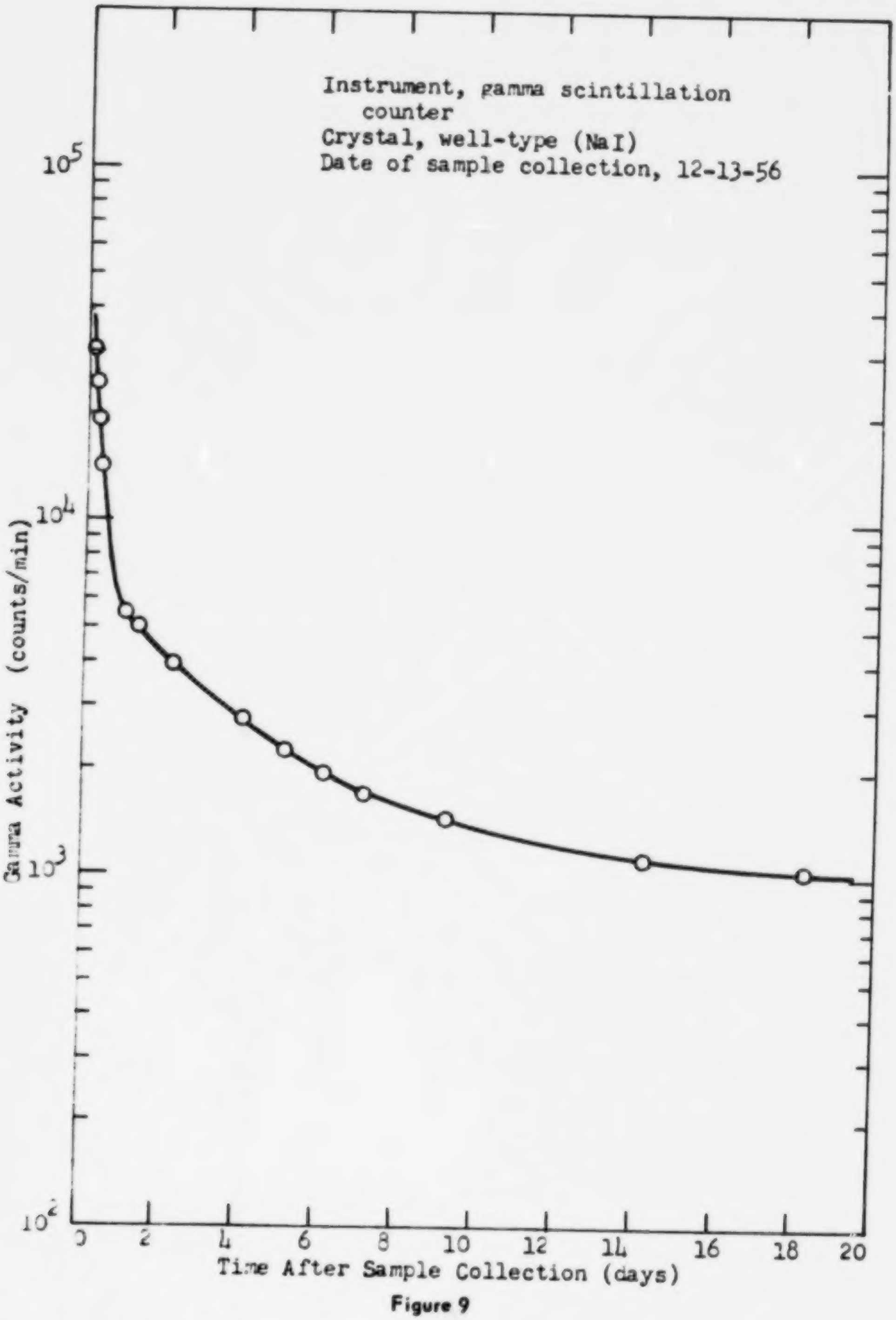

Figure 9

Gross gamma decay for reactor cooling water: Basic sulfide group.

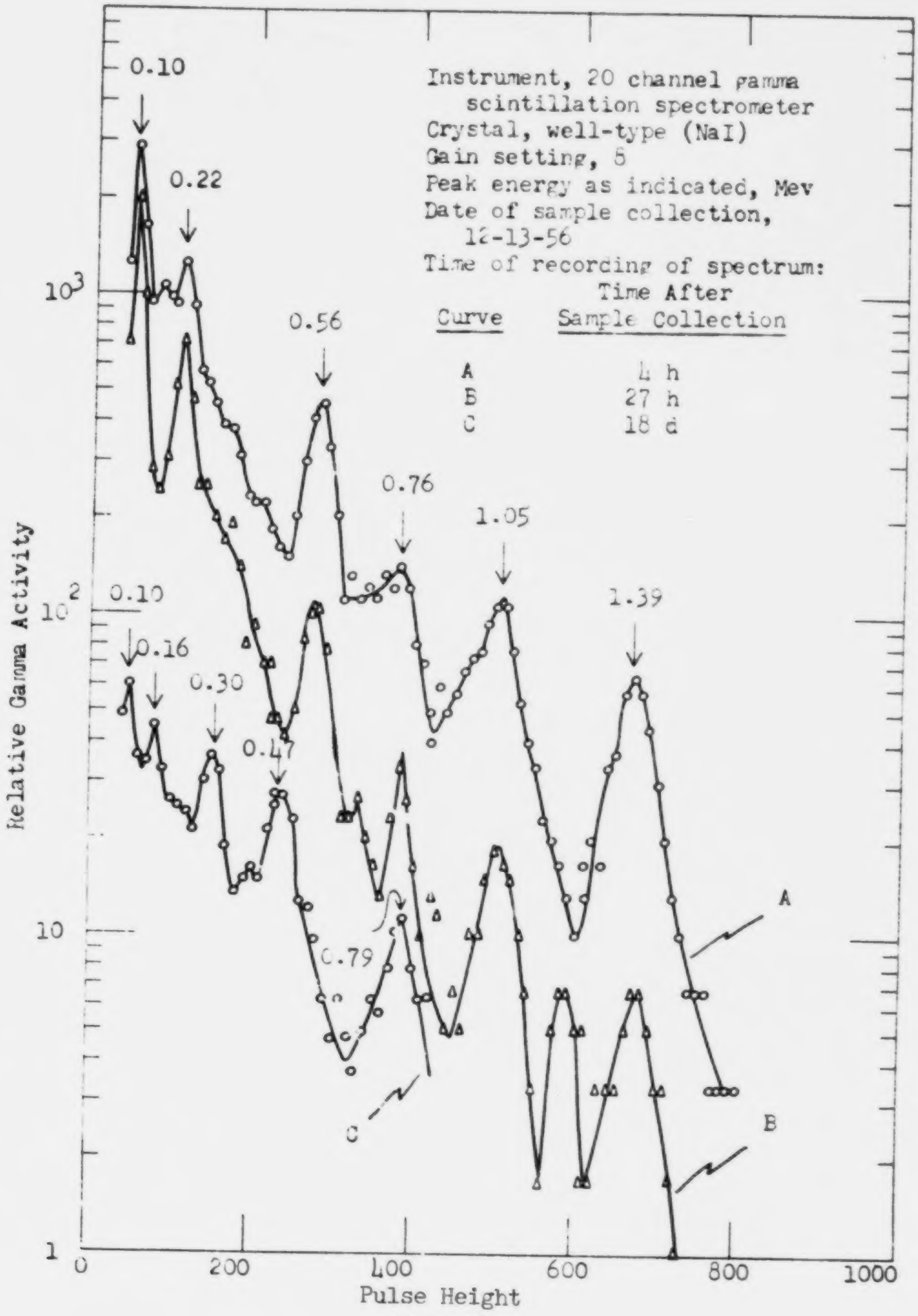

Figure 10

Gamma spectral data for reactor cooling water: Alkaline earth group.

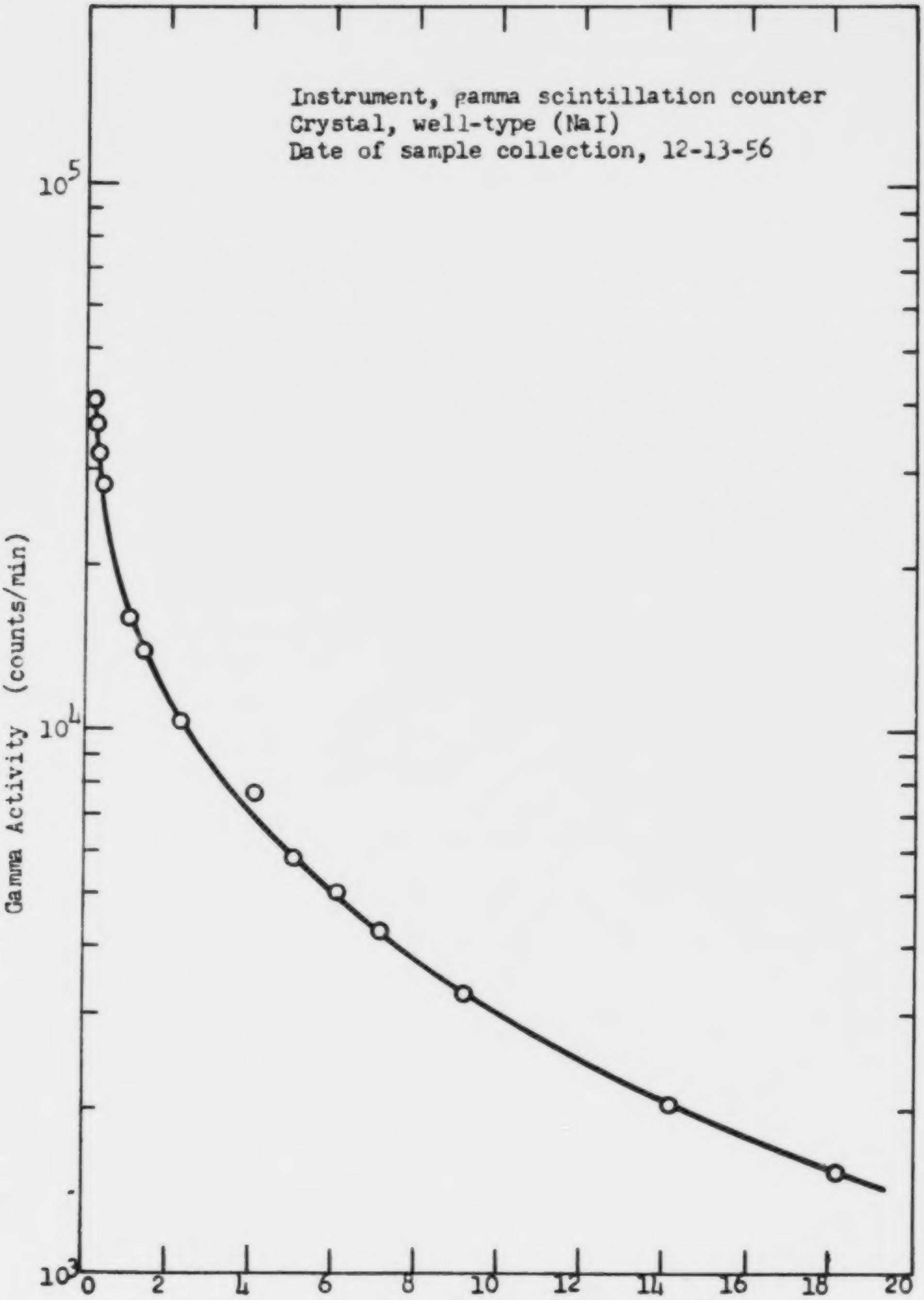

Figure 11

Gross gamma decay for reactor cooling water: Alkaline earth group.

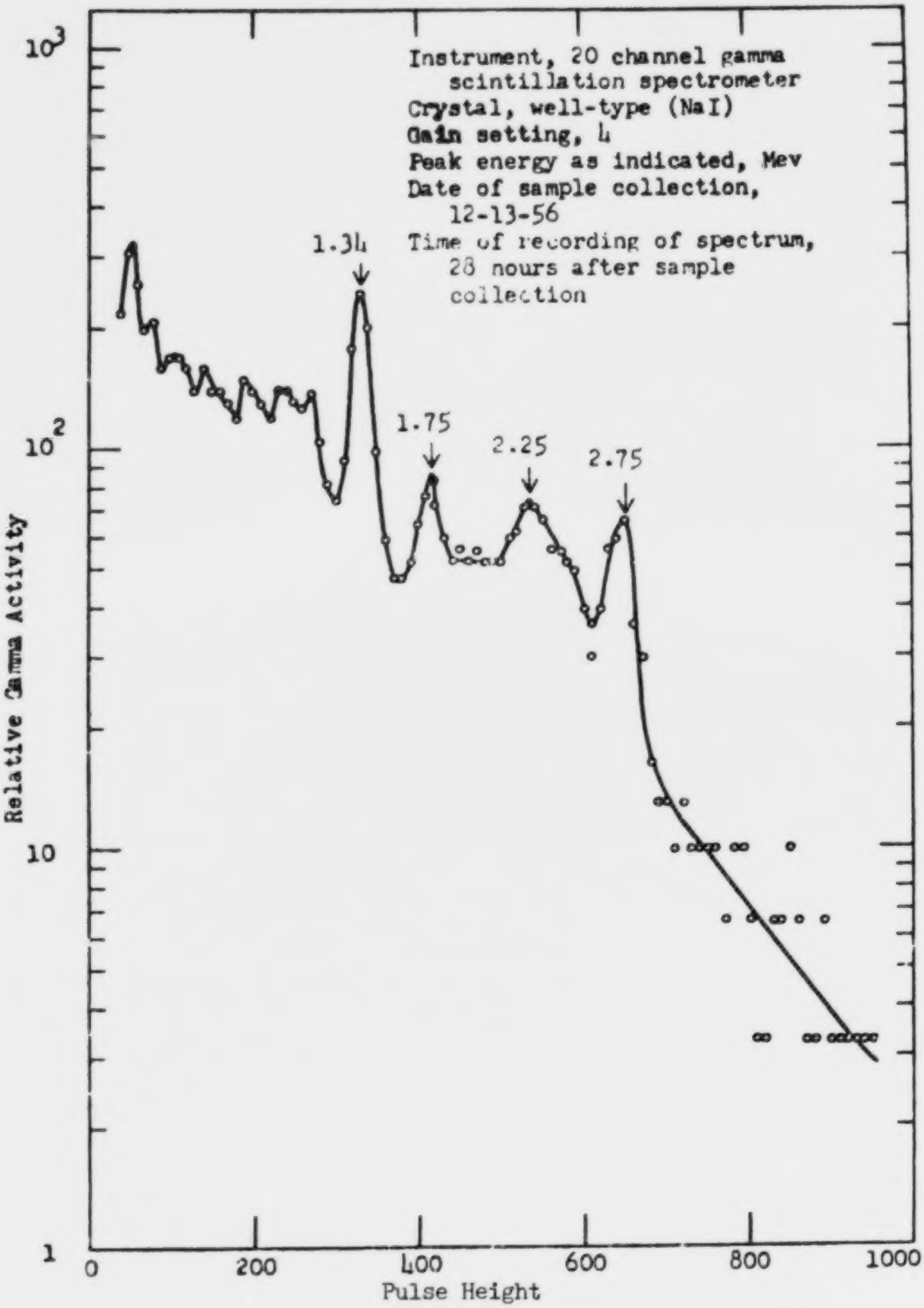

Figure 12

Gamma spectral data for reactor cooling water: Soluble group.

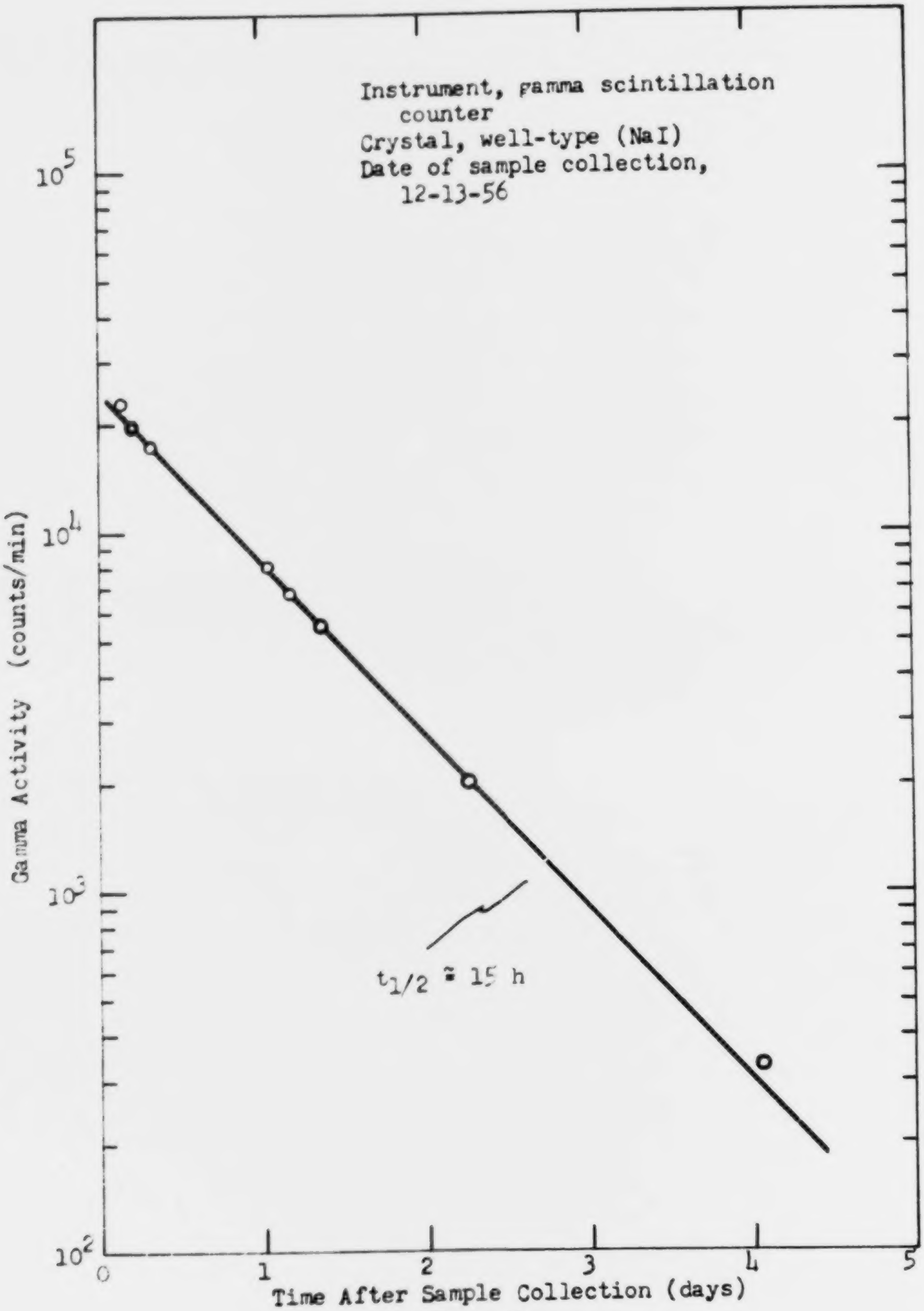

Figure 13

Gross gamma decay for reactor cooling water: Soluble group.

Table II

RADIONUCLIDES IDENTIFIED IN REACTOR COOLING WATER

Date of Sample Collection, 12/13/56

Chemical Group	Radioactive Nuclide	Half Life	Estimated Concentration (μc/ml)*	
Silver	I^{131}	8.05 d	$8x10^{-5}$	$3x10^{-3}$
	$\to Xe^{131m}$	4.8×10^{-10} s		
	I^{132}	2.3 h		
	I^{133}	21 h	$6x10^{-4}$	
	$\to Xe^{133}$	5.27 d		
	$\to Cs^{133m}$	6.0×10^{-9} s		
	I^{135}	6.7 h		
	$\to Xe^{135}$	9.2 h		
	$\to Cs^{135m}$	2.8×10^{-10} s		
Acid Sulfide	Cu^{64}	12.8 h	1×10^{-3}	
	Tc^{99m}	6.0 h	7×10^{-4}	
Hydroxide	Np^{239}	2.33 d	5×10^{-4}	
	Mn^{56}	2.58 h		
	Na^{24}	15.0 h		
	Y^{91m}	50 m	1×10^{-4}	
Basic Sulfide	Np^{239}	2.33 d	1×10^{-4}	
	Mn^{56}	2.58 h	7×10^{-4}	
	Co^{60}	5.2 y	1×10^{-5}	
Alkaline Earth	Np^{239}	233 d	2×10^{-4}	
	Sr^{91}	9.7 h	[illegible] $\times 10^{-4}$	
	$\to Y^{91m}$	50 m		
	Pa^{233}	27.4 d	1×10^{-5}	
	Ba^{140}	12.8 d	1×10^{-5}	
	$\to La^{140}$	40.2 h		
Soluble	Na^{24}	15.0 h	3×10^{-3}	

---

* At time of sample collection.

12.8 d Ba^{140}. Barium-140 is confirmed by the presence of the 0.47 and 0.79 Mev peaks of its daughter 40.2 h La^{140}. Decay data for this group are shown in Figure 11.

Gamma spectral and decay data for the Soluble Group, as shown in Figure 12 and 13, showed the predominant source of activity in this group to be 15.0 h Na^{24}. The 1.34 and 2.75 Mev peaks are representative of this nuclide. The other two peaks shown on the spectral curve are double and single escape peaks resulting from pair production by the higher energy photon(5).

A summary of radioactive nuclides identified in the reactor cooling water sample as well as an estimate of their concentrations is given in Table II. The quantitative data are meant to illustrate orders of magnitude only, being calculated using an assumed average chemical yield of 75% and gamma counting efficiency of 33%.

V. DISCUSSION

The gamma scintillation spectrometer makes possible in many instances rapid identification and quantitative determination of gamma emitting radionuclides. The method simplifies radiochemical analysis but does not replace it.(1)

Possible applications of the technique in sanitary engineering and public health are many. It has been shown to be useful in analysis of fission products in liquid wastes and reactor cooling water, in the activation analysis of drinking water, and in the identification and quantitative determination of radionuclides in air and water at or near background levels.

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Assay of Radionuclides: Beta Emitters

I. INTRODUCTION

By taking advantage of the radioactive properties of many nuclides, it is possible to detect their presence or measure their quantity in amounts far smaller than with ordinary chemical or physical methods. For example, one milliliter of a solution containing 10^{-12} mg/ml of sodium-24 would give approximately 2,000 c/m on the usual laboratory G-M counter. This readily detectable quantity corresponds to only 10^{-9} parts per million of sodium. Such trace determinations are far beyond even flame or arc spectroscopy.

Although the assay of many radionuclides can be readily accomplished utilizing beta and gamma spectrometers, the present lecture will be limited to radiochemical analyses which are practicable using the ordinary laboratory G-M counter.

II. IDENTIFICATION OF BETA EMITTERS

In elementary radiochemical assays, there are two convenient "handles" by which unknown nuclides may be grasped and identified. These are (a) half-life and (b) maximum beta energy. Often the easier of the two to measure is the half-life; and if the half-life is known to within 10 or 20 percent, many times the identity of the emitter can be established.

A. Half-Life Determination

The technique for measuring the half-life is basically very simple: count the sample once, then once again under identical conditions when the count has dropped to one-half its original value. The time elapsed between the two counts is equal to the half-life. Such a procedure would be, in general, rather tedious. Therefore, the usual procedure is to make the second measurement after any arbitrary length of time (T), and then determine the half-life ($T_{1/2}$) using the decay equation:

$$\frac{A_t}{A_0} = e^{-\frac{0.693\,t}{T_{1/2}}}$$

Another approach is to plot a series of measurements on semilogarithmic graph paper where the above equation becomes a straight line (see Figure 1). In the absence of semi-log paper, the log coordinates may be ruled off using the "C" scale of the slide rule.

Still a third approach is to read directly the number of half-lives represented by the time interval as a function of percentage count decrease from the "Radioactive Decay, Semi-log Plot" graphs of the Radiological Health Handbook.

Difficulty is encountered in utilizing these techniques for the determination of either very short or very long half-lives; a substance such as ^{28}Al which has a 2.3 minute half-life might represent the short side for usefulness, while for a nuclide such as ^{226}Ra, the line would show a slope of only a few percent in a hundred years. Nevertheless, even such a general knowledge of the half-life often will permit identification of the emitter from tabular listings of available nuclides according to half-life, such as the tables, "Available Radioisotopes Listed According to Half-Life," in the Radiological Health Handbook.

The straight-line semi-log plot of activity vs. time shown in Figure 1 results only when the activity being counted is from but a single

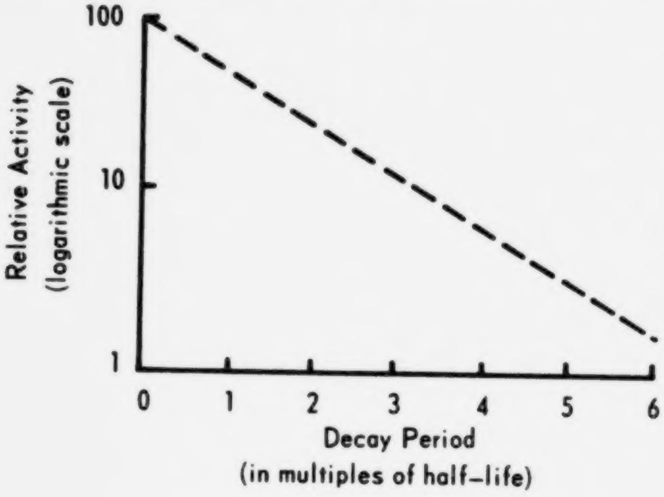


Figure 1. Radioactive Decay

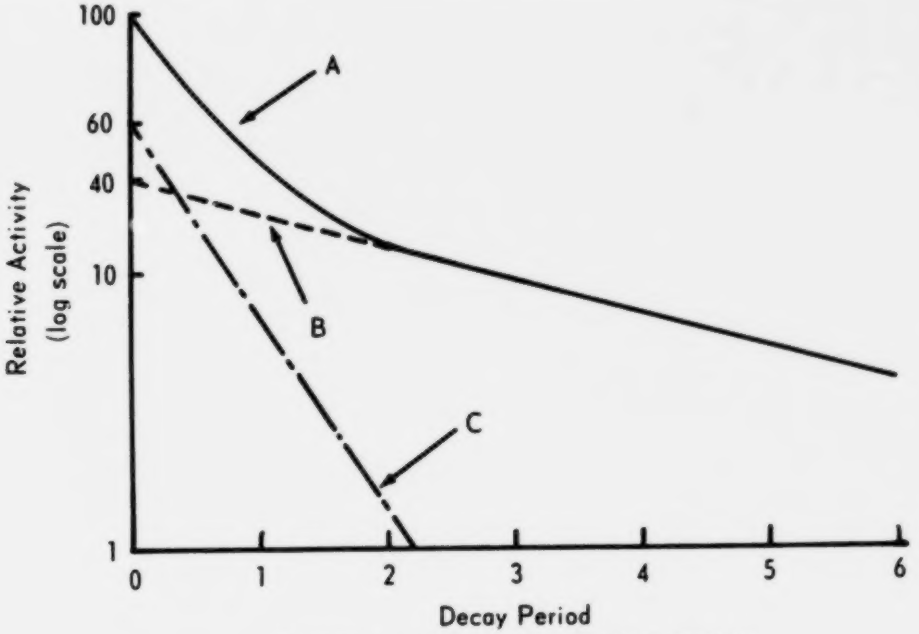
radionuclide. If more than one emitter is present, and the half-life of the second differs significantly from the first, the semi-log plot will more closely resemble Figure 2. In such a case, the straight line which results after the short-lived component has decayed away may be used to determine the longer half-life then extrapolated back to the initial time and its contribution subtracted from the total curve, point by point. The steeper net curve which results may then be used to find the half-life of the short-lived component. This method may be used with success to disentangle up to three or four emitters observed in combination.

B. Measurement of the Beta Maximum Energy

If the half-life should prove to be insufficient to establish, unambiguously, the identity of the nuclide, then the energy of the emission should be measured. For any nuclide which emits a mixture of betas and gammas, or only gamma rays, the most elegant method of identification is, undoubtedly, the gamma spectrometer. However, if the nuclide is a pure beta emitter, or in the case of a beta-gamma emitter if no gamma spectrometer is available, the procedure is as follows:

1. Set up the sample to give a convenient counting rate (~10,000 c/m) in a conventional laboratory G-M counter.
2. Insert aluminum absorbers of known density-thickness (mg/cm^2) between sample and G-M window. Density-thickness (mg/cm^2) is preferable to thickness (cm) as a unit of measurement because it allows interchanging of absorbers of various materials without reference to atomic number or density.
3. Plot c/m vs. mg/cm^2 on semi-log paper with c/m on the log axis.
4. From this plot, determine the aluminum half-thickness, i.e., the thickness of aluminum required to reduce the count rate by one-half.
5. Obtain the maximum energy of the beta rays from an empirical plot of energy vs. aluminum half-thickness for many different emitters; such a plot is shown in Figure 4.

The plot of absorber vs. activity will usually approximate a straight line for a single nuclide which is a pure beta-emitter (see Figure 3a). On occasion, however, an unknown nuclide will result in a curve such as is shown in Figure 3b.



CURVE A – COMPOSITE DECAY CURVE
CURVE B – LONGER-LIVED COMPONENT
CURVE C – SHORTER-LIVED COMPONENT

Figure 2. Radioactive Decay

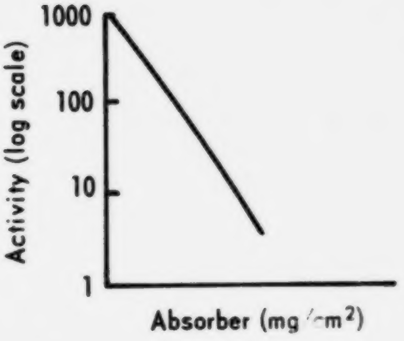


Figure 3a. Absorption [illegible]e

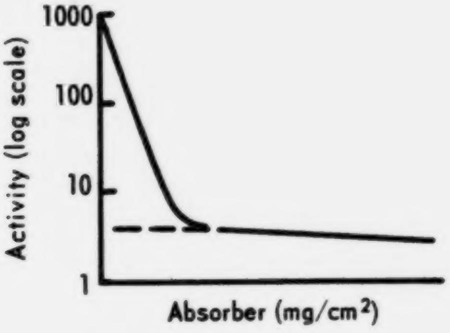


Figure 3b. Absorption Curve

The constant portion of the curve indicates the presence of gamma rays which are not appreciably attentuated by the aluminum absorber. In such a case, the constant gamma value should be subtracted out of the total curve, point by point, and the more nearly straight line which results is then used for the determination of half-thickness.

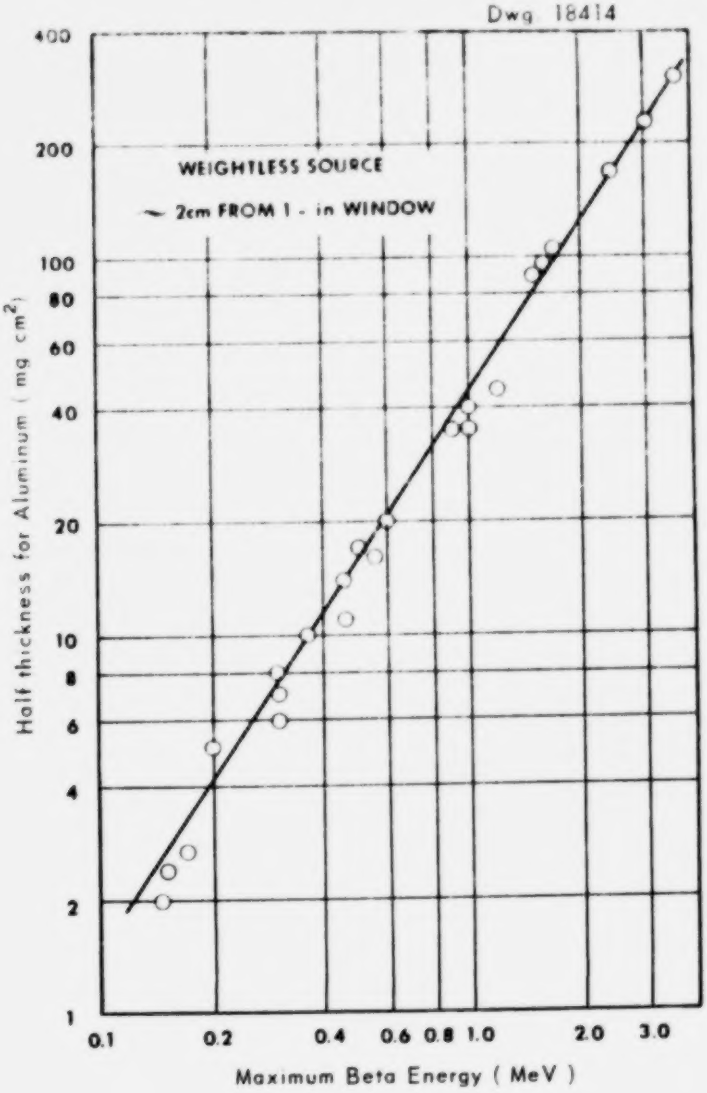


Figure 4. Beta Half-Thickness vs. Energy

III. CHEMICAL SEPARATION TECHNIQUES

Many times a given sample will contain a mixture of radionuclides such that half-life and/or beta-energy determinations do not lead to ready identification. In these circumstances, techniques for separation of the various radionuclides may have to be employed. Following are discussions of some of these techniques.

A. Carrier Techniques

Ultra-low concentration chemistry (often 10^{-10} molar) is beset with a number of pitfalls which are lacking in chemical practice on a larger scale. Entire quantities of the order of 10^{-14} grams in a few milliliters of solution can readily be adsorbed onto the glass walls of the containing vessel. Peculiar colloidal properties are exhibited, and equilibria become quite concentration-dependent. These difficulties may be avoided by adding "carrier" materials to the radioelement samples. From 0.1 to 20 milligrams of the ordinary stable form of the element may be added to the radioactive sample, whereafter the quantities of precipitates, etc., are no longer sub-microscopic, and the chemistry becomes more normal. For example, if one were attempting to separate barium from the mixture barium-lanthanum, ordinary $BaCl_2$ could be added with the result that the radioactive barium is carried down during precipitation of the stable barium. If the element to be separated is a rare one, so that no stable carrier is readily available, then often a non-isotopic carrier composed of some element with similar chemical behavior may be used. For example, it has been found that LaF_3 will carry ^{239}Pu along with it in analyses. Often, however, such carrier tricks will work only for one valence state of the carrier so that oxidation-reduction reactions must be avoided or carefully scrutinized.

B. Precipitation

Precipitation is a familiar chemical separation practice and not much need be said about it here. If it is desired to separate silver out of silver-copper mixture in HNO_3, then HCl would precipitate out the silver as AgCl. Further, purification could be made with repeated dissolution-precipitations between NH_4OH and HCl.

C. Distillation

Distillation is sometimes useful but almost always hazardous. RuO_4 is distillable, for example.

D. Liquid-Liquid (or "Solvent") Extraction

Solvent-extraction techniques are perhaps less well known but very useful. Three cases might be cited.

(1) Extraction of an organic-soluble component, such as free iodine in water extracted by chloroform or carbon tetrachloride,

(2) Extraction of a complex, such as ferric chloride from 8M HCl with ether,

(3) Direct extraction from solution with an organic complex-former, such as zirconium with thenoyltrifluoroacetone in benzene, or uranium with tributyl phosphate.

E. Electrodeposition

Electrodeposition is excellent for obtaining metals in very pure form, as for example, plating out radio-silver directly onto a platinum counting disc.

F. Ion Exchange

Ion exchange is perhaps the newest and most useful chemical separation technique. The ion exchange materials are organic resins which exchange their metal ions, generally, for metal ions having higher valences: Na^+ for Ca^{++} for Al^{+++}. They have been applied in household water softeners using Zeolite which exchanges its sodium ions for the water's calcium ions, thus making hard water soft. Usually a column of specified size is filled with the resin and the solution to be separated permitted to run through at a specified slow rate. The components will be left in the column in order of their ascending valence. The column may then be physically sectioned, or each valence flushed out with increasing concentrations of HCl.

After utilizing the various chemical separations which might be required, identification of the nuclide may be made using the techniques previously outlined.

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Basic Principles of Nuclear Reactors

I. INTRODUCTION

On December 2, 1942, the world's first nuclear reactor went critical. This scientific achievement signaled the beginning of a new era: the atomic age. Utilizing the energy liberated from the heart of the atom, mankind's dream of an unlimited source of power was realized; his dependency on fossil fuel, a diminishing supply, was broken.

Scientific progress and technological advances in the design and construction of reactors have made the cost of producing electrical power by nuclear reactors competitive with production by conventional methods in many areas. Advances in reactor design coupled with declining resources of fossil fuels will undoubtedly result in even more extensive use of nuclear energy for electrical power generation in the future.

II. FISSION

Fission is the process whereby an atomic nucleus splits, usually as a result of a neutron-induced interaction, into two lighter nuclei called "fission fragments." The neutrons serve as projectiles to add energy to the nucleus, upsetting its energy balance. Since neutrons are uncharged particles, they are not affected by the coulomb field of the atom's nucleus, and can enter the nucleus more readily than can charged particles. Simultaneously with the splitting, additional neutrons, gamma rays, and a quantity of energy are liberated. The two or three neutrons released during the fission process can enter other fissionable nuclei, causing additional fissions--a "chain reaction."

The amount of energy released per fissioning nucleus can be calculated from Einstein's formula, $E = mc^2$ (energy equals mass times the speed of light squared). The difference in mass of the fission products and that of the original atom is the "mass difference" which has been converted into energy during the fission process.

Almost all (over 99 percent) the neutrons produced in the fissioning process are released within 10^{-14} seconds after fission occurs; these are known as "prompt neutrons." About 0.64 percent of the neutrons produced are expelled from the fission products over a period of 0.05 to 55.6 seconds; these are called "delayed neutrons."

Even though the delayed neutrons comprise a very small part of the neutrons released, they have an important bearing on the time-dependent behavior of reactors. However, in a discussion of the rudimentary principles of operation of a reactor, only the prompt neutrons need be considered.

These prompt neutrons, emitted in the fission process, range in energy from nearly zero to about 10 MeV, with the majority having energies of about 1 or 2 MeV.

There are several nuclides which are fissionable by neutrons of all energies and can sustain a chain reaction. Of these fissionable or "fissile" materials, only ^{235}U is both naturally occurring and sufficiently long-lived to be of practical value as a nuclear fuel. This fact alone makes uranium of prime interest for reactor use. However, ^{235}U constitutes only about 0.72 percent of natural uranium; ^{238}U, which constitutes over 99 percent of natural uranium and is not a fissile material (although it can be fissioned by fast neutrons, it will not alone sustain a chain reaction), is important nonetheless as the raw material for the production of ^{239}Pu. As will be discussed later, this fact is central to the concept of breeder reactors.

III. REACTOR THEORY

To achieve a self-sustaining nuclear reaction, or criticality, the following condition must prevail: The rate of neutron production must equal the rate of neutron losses. The isotope ^{235}U is primarily responsible for the fission process in a reactor. Uranium-238 will only fission with neutrons having energies greater than 1 MeV. Even then, the probability that a fast neutron will fission ^{238}U is relatively low. Fast fission of ^{238}U accounts for only a few percent of the total fissions.

All neutrons produced in the fission process are not available to sustain the process. Some diffuse away from the fuel region and are lost. Some that remain in the fuel region undergo non-fission capture by the fuel and the materials used for reactor construction.

Production and losses are illustrated by the neutron cycle in Figure 1. Upon the fissioning of ^{235}U, the fast neutrons released may cause ^{238}U atoms to fission, producing additional neutrons, or the fast neutrons may strike the moderator. The moderator reduces the neutrons' energy through elastic scattering to thermal or near thermal energies. At these energies they are most likely to cause further fissioning of ^{235}U.

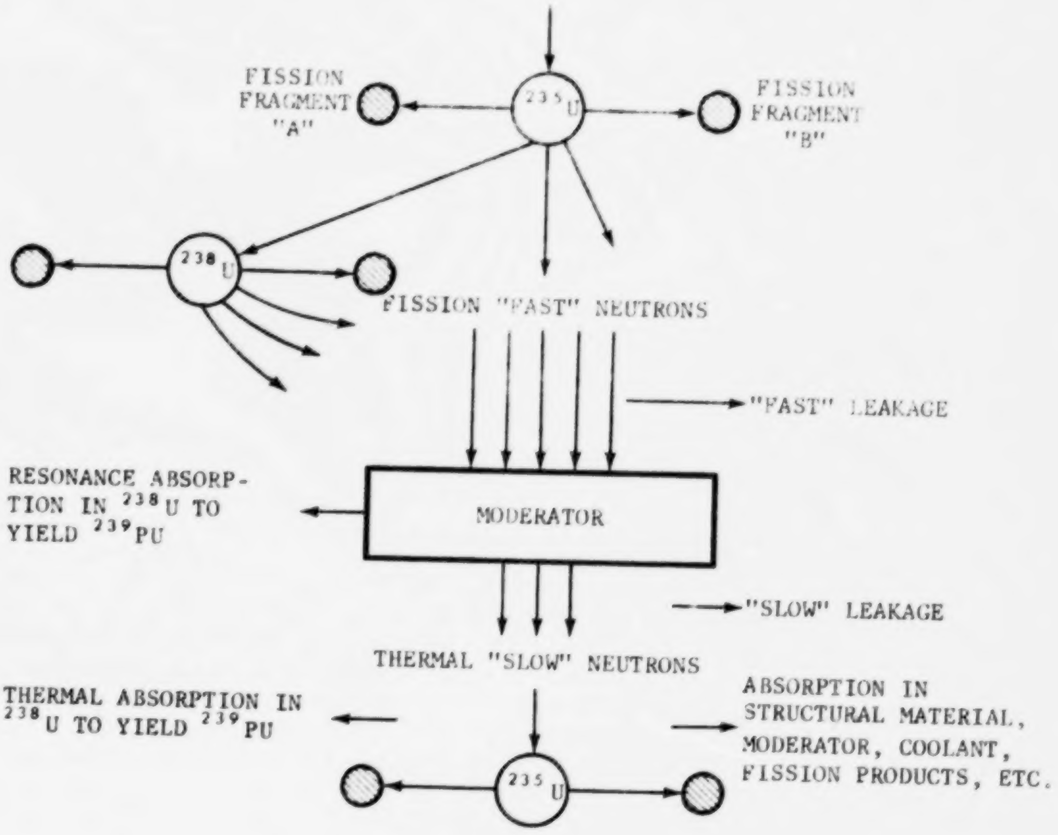

FISSION FRAGMENT "A"
235 U
FISSION FRAGMENT "B"
238 U
FISSION "FAST" NEUTRONS
"FAST" LEAKAGE
RESONANCE ABSORPTION IN 238U TO YIELD 239PU
MODERATOR
"SLOW" LEAKAGE
THERMAL "SLOW" NEUTRONS
THERMAL ABSORPTION IN 238U TO YIELD 239PU
ABSORPTION IN STRUCTURAL MATERIAL, MODERATOR, COOLANT, FISSION PRODUCTS, ETC.
235 U

Figure 1.--The Neutron Cycle

(Thermal Reactor Fueled with ^{235}U and ^{238}U)

Adapted from Kaplan, I., "Nuclear Physics," 1963

To reduce the loss of neutrons escaping from the system, one can increase the size of the system. The minimum quantity of fissile material necessary to maintain a chain reaction is called the "critical mass." The critical mass for a given reactor depends on a wide range of factors, although for a specific reactor design it always has a definite value.

The ratio of neutrons available for fissioning in any one generation to the number available in the immediately preceding generation is designated as the effective multiplication factor, k_{eff}.

$$k_{eff} = \frac{{}^{1}_{0}n_{g}}{{}^{1}_{0}n_{g-1}}$$

Here:

k_{eff} = k effective

${}^{1}_{0}n_{g}$ = neutrons produced in a generation

${}^{1}_{0}n_{g-1}$ = neutrons produced in previous generation.

For the chain reaction to be sustained in a steady state, k_{eff} must be equal to one. The system is critical, and the number of neutrons available for further fissioning balances those lost through leakage or capture. When k_{eff} is less than one, the chain reaction is not maintained; the system is subcritical. More neutrons are lost than are being produced. For k_{eff} greater than one, a surplus of neutrons is being produced in each generation; the system is supercritical.

IV. REACTOR DESIGN

A. Core

The core consists principally of the fuel, moderator, and structural material.

Because of design and operating difficulties inherent in fuels consisting of naturally occurring uranium, enriched uranium--with a higher percentage of ^{235}U than the 0.72 percent occurring naturally--is used. The degree of enrichment depends on the design features of the reactor, or vice versa. Enriched fuel has several advantages, including:

1. A wider variety of materials (coolant, moderator, construction material, etc.) is possible in the design of the reactor.
2. High fuel burnup is achieved; that is, a larger percentage of the ^{235}U nuclei are fissioned, so the fuel can be more efficiently used.
3. Higher power densities (power output per core volume) can be obtained with enriched than with natural uranium.

The fuel is formed into rods or plates to improve heat removal. It is encased in a protective cladding which contains the fission products, prevents chemical reactions between fuel and moderator, and provides structural support. Several physical factors should be considered for cladding materials: low neutron capture probability, structural strength at high temperatures, good heat transfer, and non-corrosive characteristics. Commonly used cladding materials that meet these requirements are aluminum, zirconium, alloys of the two, and stainless steel.

B. Moderator and Reflector

Slow neutrons have a higher probability of producing fission in ^{235}U than fast neutrons. Since neutrons produced by fission have a wide spectrum of energies, a moderator is used to slow them down and enhance the fission process. On the average, neutrons lose more energy per elastic collision with practicles of equal mass (e.g., hydrogen nuclei) than they do in colliding with heavier particles. For this reason, materials with low atomic weight--generally hydrogen or hydrogenous compounds--are used for moderators.

Because some neutrons are scattered at angles greater than 90 degrees (i.e., in the general direction from which they came), moderating materials are also suitable as reflectors to reduce neutron leakage. Thus, material of low mass number interspersed among the fuel elements serves as a moderator and outside the reactor core serves as a reflector.

The reflector and moderator material should have a low probability for neutron capture. Materials most commonly used for reflectors and moderators are:

1. WATER

Because of its hydrogen content, water serves as a good moderator and reflector. Since water is also cheap, it is often used in reactors in a dual role as moderator-reflector and coolant. A disadvantage is its relatively low boiling point, which requires expensive high-pressure vessels and piping for high-temperature operation. Also, since water has a relatively high thermal neutron absorption cross section, the fuel in a water-moderated reactor must be enriched with ^{235}U to offset the additional neutron loss.

2. HEAVY WATER

Of all the moderators in use, heavy water (^2H_2O) has the lowest neutron absorption to thermalizing ratio. It is a very good moderator and coolant, but high production and purity-maintenance costs limit its use.

3. GRAPHITE

Graphite is easily purified, machined, and relatively inexpensive. It also has a very low thermal neutron capture cross section. Its brittleness causes difficulties when it is used as a structural material. Graphite also reacts with oxygen at high temperatures.

4. BERYLLIUM AND BERYLLIUM OXIDE

Both these materials are good moderators, but are relatively expensive. Their disadvantages are that they are brittle and react with air or moisture in air. They are also very toxic and, therefore, hazardous to fabricate. Beryllium and its oxide are only used in special applications.

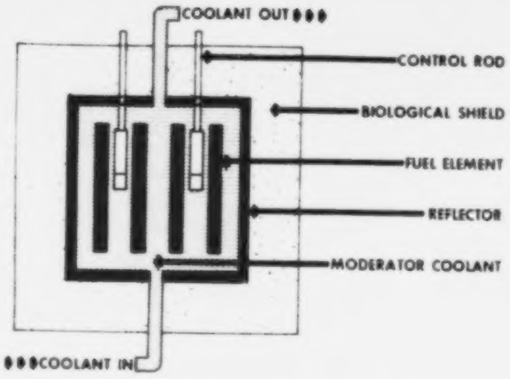


Figure 2.--Schematic Diagram of a Reactor

When the arrangement of the core is such that the fuel and the moderator are separate entities, it is called a "heterogenous reactor" (schematically shown in Figure 2). Nearly all reactors are of this type. In a few operating reactors the fuel is intimately and uniformly mixed with the moderator. These are called "homogeneous reactors."

C. Coolant

The great quantities of heat produced in the reactor core must be removed to prevent the fuel elements from melting. For electricity production, the removed heat is used to produce steam to power generators. The cooling system removes the heat by circulating a heat-absorbing material through the core. The heat generated in the fuel elements is transferred to the coolant and circulated out of the core. The most commonly used coolants are:

1. WATER AND HEAVY WATER

These are both good coolants and good moderators. The chief disadvantages are water's corrosiveness and its low boiling point. Pressurized water reactors solve the latter problem by keeping the water under high pressure, thereby raising its boiling point. Boiling water reactors--although pressurized--allow the water to boil, forming steam.

2. GASES

Air was used as the coolant in the early reactors. Now gas-cooled, high-power reactors use gases other than air, since air reacts unfavorably with structural materials in the core. Helium and carbon dioxide have been successfully used as gas coolants. Gas cooling systems require expensive pumping arrangements unless the gas is kept under high pressure.

3. LIQUID METALS

Liquid metals--particularly sodium and sodium-potassium--are used as coolants where neutron moderation is undesirable, as in fast reactors.

D. Shielding

Of the radiations emitted by fission fragments produced in the fuel elements, neutrons and gamma rays present the most serious shielding problems. Alpha and beta particles are generally absorbed by the fuel cladding and other materials used in reactor construction.

Since the probability of neutron capture (removal) increases as the neutron kinetic energy decreases, it is desirable first to moderate or slow the neutrons down and then remove them through capture reactions. Good neutron moderators are low-density materials with a high hydrogen content. Good absorbers for gamma rays are high-density materials, such as lead or iron. Concrete has proven a satisfactory compromise for shielding against both gamma rays and neutrons from reactors. It contains both low and fairly high atomic weight materials--hydrogen and silicon. In addition to its shielding properties, concrete has good structural qualities and is relatively inexpensive. Iron punchings and boron can be added to enhance gamma-ray shielding and neutron capture respectively. Because of these assets, concrete is used as a shielding material in a majority of reactors.

Water is used for a shielding only in special applications. Although its shielding properties are good, water generally presents considerable construction difficulties.

E. Control and Safety

The operation of a reactor can be described in terms of the multiplication factor, k_{eff}, mentioned previously. Control or regulating rods maintain the proper k_{eff} factor for various stages of reactor operation.

Control rods are made of materials which have a high probability of capturing neutrons, such as cadmium or boron, and can remove the neutrons from the core region, making them unavailable for further fissioning. The neutron population is controlled by moving the rods in or out of the core region. With precise positioning of these rods, the point where k_{eff} equals one is reached. A stable, critical condition then exists in the reactor.

Control rods are positioned electrically and/or hydraulically, and are classified as either coarse or shim rods. The names refer to their degree of adjustment. The former are used for making gross adjustments, while the shim rods are used for making finer adjustments in the number of fission events. Other rods, called safety or scram rods, are strategically positioned in the core. In the event of a drastic increase in k_{eff}--hence fissions--these rods are inserted in the core immediately to shut down the reactor. Control rods may also be used as safety rods.

A fairly new concept of reactor control is adding boron directly to the coolant. The concentration of boron in the coolant is varied for routine control with major reactivity changes controlled by rods.

V. REACTOR CLASSIFICATION

In addition to the reactor categories already mentioned--heterogenous-homogenous, coolant, etc.--reactors may also be classified by:

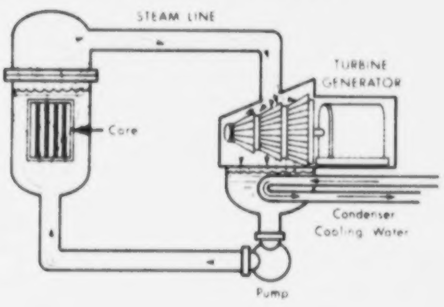
A. Use

1. Research Reactors--a reactor primarily used for research, either as a prototype or proving ground for future reactor design or operated to produce neutrons for pure scientific research.

2. Power Reactors--heat produced in the core is removed by the coolant and put through various heat-exchange systems, where it is eventually converted to electrical or mechanical energy (boiling water and pressurized water power reactor systems are schematically shown in Figure 3).

3. Isotope Production Reactors--high neutron fluxes produced in the reactor are used to produce radioisotopes.

BOILING WATER REACTORS



PRESSURIZED WATER REACTORS

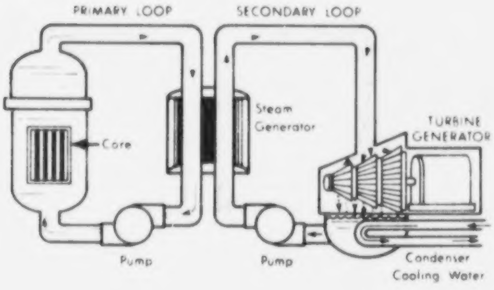


Figure 3.--Boiling and Pressurized Water Reactor

4. Breeder Reactors--the ^{238}U present in the fuel is incorporated into the reactor design. Slow neutrons are absorbed by the ^{238}U nucleus and the following reactions occur:

$$^{238}_{92}U + ^{1}_{0}n \xrightarrow{^{0}_{-1}\beta} ^{239}_{92}U \xrightarrow{^{0}_{-1}\beta} ^{239}_{93}Np \xrightarrow{^{0}_{-1}\beta} ^{239}_{94}Pu$$

Plutonium-239, like uranium-235, is fissionable (fissile) with thermal neutrons and will sustain a chain reaction. Thus, in addition to producing energy which may be used for power generation, the breeder reactor produces more reactor fuel than it consumes.

B. Neutron Energy

1. Thermal Reactors--thermal energy neutrons (<100 eV) are used to produce fission.

2. Intermediate Reactors--intermediate energy neutrons (100 eV to 0.1 MeV) are used to produce fission.

3. Fast Reactors--high energy neutrons (>0.1 MeV) are used to produce fission.

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Medical Applications of Radiation

I. INTRODUCTION

Radiation (x-ray) for diagnosis has been employed since 1896, and radiation (radium) has been used for therapy since 1898.

The use of radiation in the healing arts has brought about great advances in health to the population of the nation. The diagnosis and treatment of disease, and research in cause, life-history and cure of disease through the use of radiation have greatly increased our life-span, our health and our happiness. However, in all cases where radiation is used, the benefits to be derived from its use must be weighed against the possible harm that may occur from the radiation exposure.

II. MAGNITUDE OF THE POTENTIAL EXPOSURE

Medical (including dental) x-ray exposure constitutes, at present, the largest source of external man-made radiation exposure to the public in the United States.

A comparison of the relative magnitudes of diagnostic x-ray exposures with the exposure levels commonly associated with nuclear energy reveals some interesting facts; among these is that dose levels involved in certain diagnostic x-ray procedures frequently are of the same order of magnitude as emergency exposure levels in the nuclear energy field.

The average 30-year gonadal exposures from various sources to the U.S. population was estimated in a report by the National Academy of Sciences and later revised as follows:

Background	3.0 roentgens
Medical x-rays (including dental)	4.6 roentgens
Weapons testing (fallout)	0.02 to 0.4 r (average 0.1 r)

As of 1954 there were estimated to be more than 160,000 x-ray machines utilized in the healing arts. This number did not include the mobile chest survey equipment.

Estimates from various sources are in reasonable agreement and indicate that there are approximately 38 million roentgenographic examinations annually of which about 32 million are made by radiologists or in hospitals, there are about 13 million fluoroscopic examinations per year, and there are about 3 million photofluorographic examinations in hospitals. In addition, in 1956 there were over 17,000,000 chest survey photofluorographs and more than 96,000,000 dental radiographs.

The dosages ordinarily employed in diagnostic x-ray are not great enough, except in a few cases, to cause acute radiation injury. The primary problem of radiation effects in diagnostic x-ray work is in the field of the chronic or long-term effects. These chronic or long-term effects are generally considered in three groups, carcinogenesis, life-span shortening, and genetic changes. In most cases, the dosages related in this discussion will be those to the gonadal areas.

III. X-RAY AND FLUOROSCOPY IN MEDICAL DIAGNOSIS

In hospitals the ratio of diagnostic to therapeutic x-ray machines is 7 to 3.

The benefit to be derived from the x-ray examination should be weighed in each case against the possible harm that may result from the radiation. Because a patient has had several x-rays in the past does not mean that essential x-rays should be withheld. The use of x-rays should be evaluated in the light of the medical need. It is clear that only the attending practitioner is in a position to decide whether a given x-ray is necessary.

A. Medical

The diagnostic applications of radiation are innumerable, there are many new uses devised each year. Through the use of appropriate techniques we can visualize not only the bones and teeth, but also the various body cavities and tissues. The use of the x-ray has greatly aided us in probing the "hidden parts" of the body for the possibility of lurking disease.

The radiation exposure from medical diagnostic x-rays (and fluoroscopes) will vary from procedure to procedure, from institution to institution, and even the same procedure on the same machine may not give the same exposure on consecutive examinations.

The use of super-voltage radiography dates back to 1945 when the first chest roentgenogram at 1,000 kv was made. Super-voltage radiography is the preparation of roentgenograms with rays generated at or above 250 kvp. This technique facilitates the visualization of air-containing structures. The lungs, components of the mediastinum, pharynx, and larynx can be more clearly outlined, and abnormalities of these parts might thus be detected at an earlier stage and demonstrated with greater clarity than would be possible with conventional roentgenograms. The use of super-voltage roentgenography to enhance conventional methods of roentgen diagnosis has been shown to be technically practical and clinically worthwhile.

The fluoroscope can be potentially the most dangerous of common x-ray uses, if not properly controlled. It can be potentially dangerous not only to the patient but also to the examiner.

NBS Handbook 60 sets certain standards for tube-table distance, shielding, coning, filtering, and cumulative timer for the fluoroscope. In addition, it states that "For routine fluoroscope the dose rate measured at the panel or table top shall be less than 10 r/min".

Handbook 60, in addition to recommendations for fluoroscopy, also makes recommendations for the x-ray equipment, the room housing the equipment, and patient and operator safety.

Presently available techniques and equipment will reduce radiation exposure from diagnostic x-ray as much as 50 to 75%. Among these are high voltage, filtered beam, beam restricting (coning), employment of fast film and screens, and full development of films.

In addition, certain other measures such as adequate tube housing, shielding of body parts not to be exposed, and other precautions on an individual basis can be taken. More elaborate equipment such as mirror-optics in photofluorography and electronic image intensifiers also will reduce unnecessary exposure.

B. Dental

Nolan (1952) and Fixott (1956) have found that a considerable number of dental x-ray units are being operated in an unsafe manner. Patients were receiving thyroid dosages (depth dose) which markedly exceeded the thyroid dosages calculated for the Windscale accident. These were on surveys where the "full mouth" series was done.

The usual "full mouth" series consists of 10, 14, or 28 individual films depending upon which institution is doing the survey and also upon the specific problems involved.

By proper application of filters, cones, and fast film it was found that the skin dosages could be reduced by 10 times or more.

IV. RADIONUCLIDES IN MEDICAL DIAGNOSIS

Hundreds of hospitals now use radionuclides as part of their diagnostic equipment. However, this field is still young, and there is much room for future development.

In general there are three categories of diagnostic uses of radionuclides: 1) Dilution techniques; 2) Flow or diffusion measurements; 3) Biochemical concentration.

A. Dilution Techniques

1. Blood Volume: - I-131 labeled human serum albumin is injected into the blood stream. After a suitable time (10 to 15 min) an aliquot of blood is removed and the amount of activity in the sample is compared with the total amount injected. The dilution gives a measure of the volume.

2. Water Volume: - Can be measured in a similar manner to blood volume, but tritium or deuterium labeled water is ingested (not injected). This dilution technique can give a measure of total body water.

3. Extracellular body water: - Can be measured in a similar manner using Na-24, or Br-82 as tracers.

4. Red cell masses: - Can be determined by using Fe-59 or Cr-51 labeled cells. The Fe-59 labeling must be done in the human body, and thus requires a donor. Cr-51 can be done on a specimen of the patient's own cells.

5. Other radionuclides useful in this technique are P-32 and C-14 glycin.

B. Flow or Diffusion Measurements

1. Cardiac output: - A given amount of iodinated serum albumin, or of Na-24 labeled material, is injected into one arm. A continuous monitoring of an artery distal to the heart can record the time for circulation and also the amount of blood pumped in a given time, by recording time and amount of radioactivity.

2. Peripheral Vascular Disorders: - Much the same techniques as for cardiac output and circulation. A given amount of radioactive material is injected into one arm, or other convenient place, and the time required for it to reach another place (e. g. the opposite arm) is a measure of the circulation time. Vascular constriction and consequent circulation restriction can be measured in this manner. Na-24 and P-32 are used in this test.

C. Biochemical Concentration

1. Liver function tests utilizing I-131.

2. Thyroid disorders: - This is the best known radionuclide diagnostic test. I-131 is selectively concentrated by the thyroid. A known amount of I-131 is given to a patient and then after a suitable time the thyroid region is counted. This measures the percent uptake. The activity of the thyroid is compared with a reading of an equal amount (control).

3. Location and extent of malignancy: - Where thyroid cancer has metastasized to other parts of the body, a diagnostic dose of I-131 followed by whole-body scanning can frequently localize areas of thyroid cancer.

4. P-32 has been used to localize brain tumors, but an internal probe is required for this method. The use of I-131 tagged human serum albumin can be used to localize tumors not found by other means. This method is fairly accurate.

5. Cu-64, K-42 and arsenic-74 have also been used in tumor localization.

V. THERAPEUTIC RADIATION

A. Therapy with Machines

The x-rays as a therapeutic tool has been utilized for many years. The therapeutic x-ray is quite flexible. Very soft Grenz rays can be used for superficial therapy, for treatment of skin malignances. However, even when very low energy Grenz rays are used, there will still be some unwanted radiation going through the superficial layers into the body. This penetrating radiation can be avoided by the use of the proper radionuclide in place of x-rays.

More penetrating radiation is used for irradiation of deeper structures. This is the field of the super-voltage x-ray therapy unit. In order not to cause skin burns, the deep therapy is done through several "ports", or entrance areas. On some units a form of "rotational" therapy is used. This is more adaptable to the cobalt units than to x-ray.

Particle accelerators are being used in the treatment of several types of malignancy.

B. Therapy with Radionuclides

The therapeutic uses of radioisotopes are grouped under three general modes of application: (1) biochemical placement, (2) physical placement, and (3) teletherapy. A fourth, or perhaps just a subdivision of physical placement, is the placement of inactive materials in the tumor and activation of the substance with a neutron beam.

1. BIOCHEMICAL PLACEMENT

(a) Selective absorption of a material needed by a particular tissue for its special function. An example of this is the selective absorption of iodine by the thyroid.

(b) The differential turnover due to the increased use of a material in the more rapid metabolism of a particular tissue. An example of this would be the metabolism of phosphorus by the bones and blood forming elements.

2. BRACHYTHERAPY

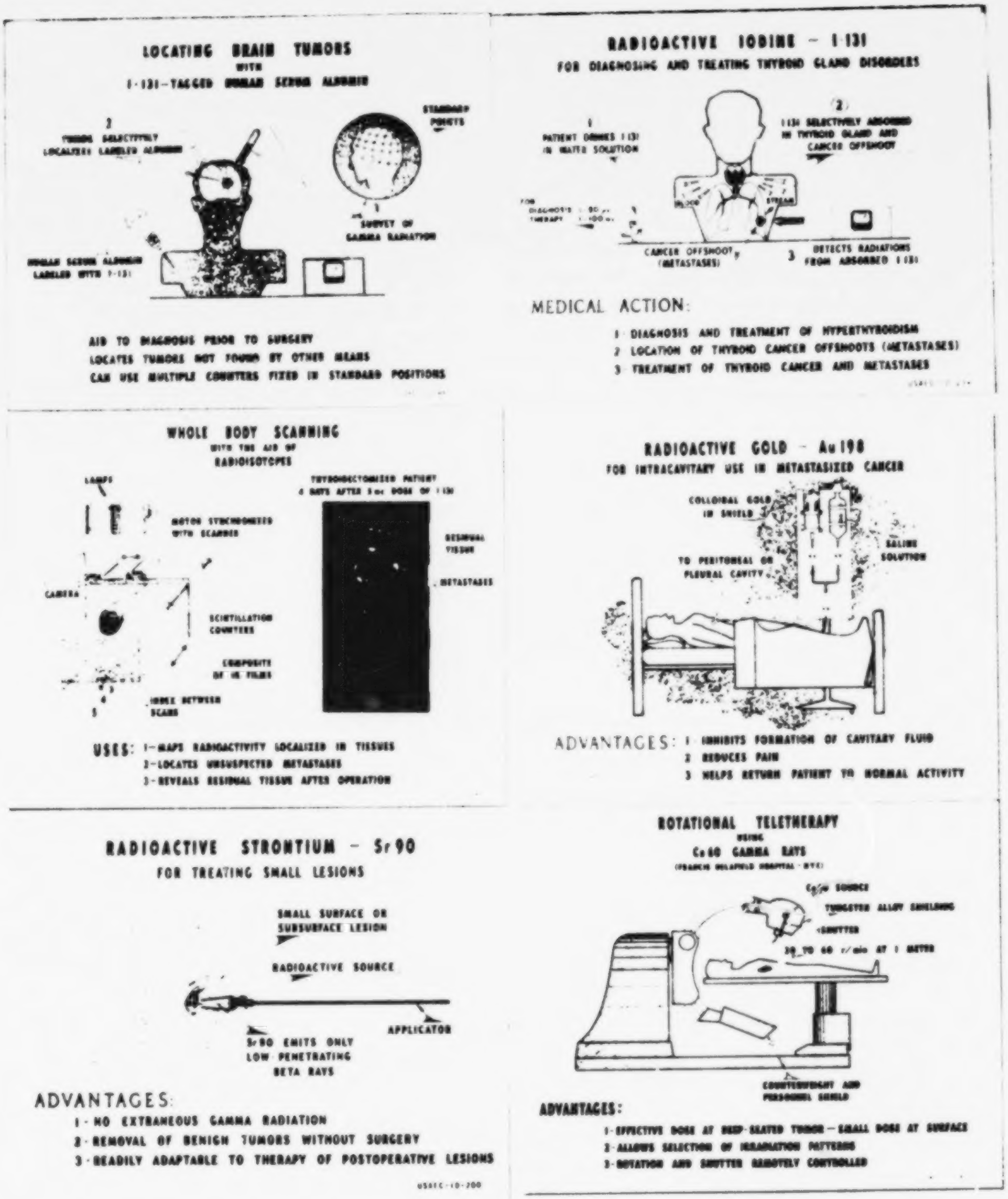
(a) As solids, such as cobalt-60, it can be used for insertion into body cavities in the same devices as have been used for a long time in radium therapy. As seeds, wire, or needles the various nuclides (Co-60, iridium-192, tantalum-182, Au-198, palladium and yttrium) can be implanted directly into the tumor to be irradiated.

(b) As liquids, (either colloids, chelates, or suspensions), in the appropriate carrier they can be used for intracavitary injection and instillation, intravenous injection, and interstitial injection. In this manner they can be used as a dispersed source, or with intravenous injection they will be transportated to inaccessible parts. A good example of the latter is the involvement of the lymphatic system in cases of cancer where the spread of the disease occurs through the lymph channels. By injecting radiogold colloid into the veins the gold reaches the lymph nodes and is there filtered out. Since the lymph nodes reacted in the same way with the cancer cells, the radiation and the part to be irradiated are in juxtaposition.

Topical application and ophthalmic applicators may be either solid or liquid, they are the radionuclides which give beta radiation so that only the superficial layers are irradiated. Sr-90 and P-32 may be used for this purpose.

3. TELETHERAPY

Co-60 is most common source. Its penetrating beam is equivalent to a 2 Mev peak x-ray unit. It is much easier to handle than is the x-ray unit as it is smaller and may be


LOCATING BRAIN TUMORS
WITH
I-131-TAGGED HUMAN SERUM ALBUMIN
2
TUMOR SELECTIVELY LOCALIZES LABELED ALBUMIN
STANDARD POINTS
SURVEY OF GAMMA RADIATION
HUMAN SERUM ALBUMIN LABELED WITH I-131
AID TO DIAGNOSIS PRIOR TO SURGERY
LOCATES TUMORS NOT FOUND BY OTHER MEANS
CAN USE MULTIPLE COUNTERS FIXED IN STANDARD POSITIONS
RADIOACTIVE IODINE – I-131
FOR DIAGNOSING AND TREATING THYROID GLAND DISORDERS
1
PATIENT DRINKS I-131 IN WATER SOLUTION
(2)
I-131 SELECTIVELY ABSORBED IN THYROID GLAND AND CANCER OFFSHOOT
BLOOD STREAM
CANCER OFFSHOOT (METASTASES)
3 DETECTS RADIATIONS FROM ABSORBED I-131
MEDICAL ACTION:
1 DIAGNOSIS AND TREATMENT OF HYPERTHYROIDISM
2 LOCATION OF THYROID CANCER OFFSHOOTS (METASTASES)
3 TREATMENT OF THYROID CANCER AND METASTASES
WHOLE BODY SCANNING
WITH THE AID OF
RADIOISOTOPES
LAMPS
MOTOR SYNCHRONIZED WITH SCANNER
CAMERA
SCINTILLATION COUNTERS
COMPOSITE OF 16 FILMS
INDEX BETWEEN SCANS
THYROIDECTOMIZED PATIENT
4 DAYS AFTER 3 mc DOSE OF I-131
RESIDUAL TISSUE
METASTASES
USES: 1–MAPS RADIOACTIVITY LOCALIZED IN TISSUES
2–LOCATES UNSUSPECTED METASTASES
3–REVEALS RESIDUAL TISSUE AFTER OPERATION
RADIOACTIVE GOLD – Au 198
FOR INTRACAVITARY USE IN METASTASIZED CANCER
COLLOIDAL GOLD IN SHIELD
SALINE SOLUTION
TO PERITONEAL OR PLEURAL CAVITY
ADVANTAGES: 1 INHIBITS FORMATION OF CAVITARY FLUID
2 REDUCES PAIN
3 HELPS RETURN PATIENT TO NORMAL ACTIVITY
RADIOACTIVE STRONTIUM – Sr 90
FOR TREATING SMALL LESIONS
SMALL SURFACE OR SUBSURFACE LESION
RADIOACTIVE SOURCE
APPLICATOR
Sr 90 EMITS ONLY LOW PENETRATING BETA RAYS
ADVANTAGES:
1 · NO EXTRANEOUS GAMMA RADIATION
2 · REMOVAL OF BENIGN TUMORS WITHOUT SURGERY
3 · READILY ADAPTABLE TO THERAPY OF POSTOPERATIVE LESIONS
USAEC-10-200
ROTATIONAL TELETHERAPY
USING
Co 60 GAMMA RAYS
(FRANCIS DELAFIELD HOSPITAL · NYC)
Co 60 SOURCE
TUNGSTEN ALLOY SHIELDING
SHUTTER
30 TO 60 r/min AT 1 METER
COUNTERWEIGHT AND PERSONNEL SHIELD
ADVANTAGES:
1 · EFFECTIVE DOSE AT DEEP-SEATED TUMOR — SMALL DOSE AT SURFACE
2 · ALLOWS SELECTION OF IRRADIATION PATTERNS
3 · ROTATION AND SHUTTER REMOTELY CONTROLLED

MEDICAL USES OF RADIONUCLIDES

FIGURE 1

rotated in many planes. There are two classes of cobalt units, the "hectocurie" units which contain only a few hundred curies, and the "kilocurie" unit which contains a few thousand curies. The hectocurie unit may be light enough to roll from place to place whereas the kilocurie unit must have a permanent installation.

Cesium-137, because of its relative longer half-life, has some advantages over cobalt-60 in teletherapy. It is of lower energy, and shielding requirements are less rigorous. However, Cs-137 is much more difficult to produce in the quantities required than is cobalt.

4. NEUTRON-CAPTURE THERAPY

Utilized in treatment of certain brain tumors. A stable isotope is made active by neutron activation after the isotope has been concentrated in the tumor through biochemical means. Boron-10 is the isotope most frequently used.

VI. RADIONUCLIDES IN MEDICAL RESEARCH

Radionuclides are being used in almost every phase of medical research, particularly the field of physiology.

All types of minerals, vitamins, and compounds that are essential to the growth and well-being of the human body have been studied by labeling particular chemical groups, or the entire molecule under study with radioactive materials such as Na-24, C-14, Fe-59, Cr-51, P-32 and others.

In this manner we can study the metabolism of calcium, sodium, phosphorus, potassium, proteins and nucleic acids.

Cholesterol metabolism is another field that is becoming more prominent, particularly in relation to cardio-vascular disease. Cholesterol metabolism can be studied using C-14 labeled materials.

Of great importance is the action of drugs upon body cells, upon the pathogenic bacteria, and upon other body chemicals, I-131, C-14, Na-24 and other isotopes are used to label drugs.

THUS WE SEE, THAT ALTHOUGH RADIATION IS A POTENTIAL HAZARD STILL IT HAS GIVEN US MANY MEDICAL ADVANCES AND PROMISES TO GIVE US EVEN MORE RAPID ADVANCEMENT.

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"Isotopes. . . An Eight-Year Summary of U.S. Distribution and Utilization" pages 7-30, 84-126. Superintendent of Documents, Washington, D.C. (March 1955).

This outline was prepared by E. Van der Smissen, M. D., Radiological Health Medical Program, Department of Special Health Services, Public Health Service, DHEW, Washington, D.C.

Industrial Applications of Radiation

I. INTRODUCTION

The purpose of this lecture is to provide general background information so that the enabling one to gain some appreciation of the many benefits derived from the uses of radiation in industry. Since all radiation uses are potentially hazardous, it will also provide a clue as to where radiation hazards might be found. It would be impractical to list here all of the specific uses to which radiation has been put. We shall discuss only the more widely used applications and those which may have some special or unique features.

II. USES OF MACHINE PRODUCED RADIATION

A. Inspection of Materials

1. RADIOGRAPHIC INSPECTION

Within a year after the discovery of x-rays, the new medium had been used to detect a defective weld. But the equipment and film available left much to be desired, and industrial radiography was used very little until 1920. Now there are thousands of industrial radiographic units in this country, in size ranging from the small portable unit used for inspection of pipeline welds, to the 25 Mev betatron for inspection of armor plate.

2. FLUOROSCOPIC INSPECTION

The fluoroscope is easily adapted to conveyorized production lines. In the past it has been limited to thin, light materials and to non-metallic goods, but with the newly developed image amplifier it can be used for heavier materials.

3. CONTINUOUS AUTOMATIC INSPECTION

(a) Thickness gauge

Continuous measurement of sheet metal, glass and rubber is performed by passing the product between x-ray tube and detector. The gauge controls the production machinery, thus accomplishing automatic operation.

(b) Height-of-fill gauge

Contains pass between an x-ray tube and a detecting element. Those which are not filled to a predetermined level permit more x-rays to pass, activating a device which removes them from the conveyor.

B. Inspection of Personnel

1. "INSPECTOSCOPE"

A whole-body fluoroscope is used to detect stolen articles, contraband, weapons, or instruments of sabotage carried by persons passing through a portal. Public health, officials and others with a knowledge of radiation effects are inclined to view this device unfavorably.

2. FLUOROSCOPIC SHOE FITTING

The shoe-fitting fluoroscope has had much appeal as a scientific curiosity and as a sales device, but its value in fitting shoes is questionable. It has fallen into disrepute as an unnecessary source of exposure, and is rapidly vanishing from the shoe stores.

C. Analysis

1. X-RAY DIFFRACTION

Crystals diffract x-rays and each crystalline material has its own diffraction pattern.

This provides a means for the identification and quantitative analysis of crystalline materials.

2. X-RAY ABSORPTION PHOTOMETRY

This technique utilizes the differences in absorption of the various elements. One application is in the determination of tetraethyl lead in gasoline. It is adaptable to continuous process control.

D. Electron-Beam Processing

1. Massive doses of radiation can be delivered by electron-beam generators. One device used for this purpose is the Van de Graaf apparatus. Another is the 1 or 2 megavolt resonant transformer x-ray apparatus, modified by removing the target and terminating the tube with a thin stainless steel window to permit electrons to leave the tube. Some of the uses to which electron-beam processing has been put are:

(a) sterilization of foods and drugs

(b) killing insects in seeds

(c) toughening of polyethylene containers by radiation-induced cross linkage of polyethylene molecules

(d) activation of chemical reaction in petroleum processing.

III. USES OR RADIONUCLIDES

A. Naturally Occurring Radioactive Materials

1. INDUSTRIAL RADIOGRAPHY

Before the availability of reactor-produced radionuclides, radium was widely used as a source for industrial radiography. It had certain advantages over x-ray machines:

(a) It could be placed inside a hollow casting so that radiographs could be taken of all sides of the casting.

(b) It could be used for panoramic exposure, i.e., a number of objects could be placed in a circle around the source and exposed simultaneously.

(c) It took less space than an x-ray machine.

(d) It was available on a lease or rental basis.

2. SELF-LUMINOUS MATERIALS

Radium is used in luminous paints for watch dials. It was formerly used on nearly all aircraft instrument dials, but for several reasons self-luminous aircraft dials are being eliminated. Radium alone is not self-luminous, but the effect of radiation on certain materials produces visible light. These materials are called phosphors. Polonium and mesothorium are also used with phosphors in luminous paints.

3. STATIC ELIMINATION

In many industrial processes the movement of non-conducting material at high speed through machinery will generate static electricity which may constitute a fire hazard or may adversely affect the quality of the product. The static charge can be removed by producing ionized air near the charged surface. Radium and polonium have been most widely used for this purpose.

B. Reactor-Produced Radionuclides

1. INDUSTRIAL RADIOGRAPHY

Radium has been replaced by other gamma emitters which are less expensive. The most commonly used radiographic sources are Co^{60}, Ir^{192}, and Cs^{137}. They are selected according to the degree of penetration required.

2. SELF-LUMINOUS MATERIALS

Sr^{90} has been substituted for Ra in some luminous paints. A new and interesting application is the krypton switch lamp used by railroads in remote locations. A flashlight has been made which uses tritium and a phosphor as a source of light.

3. IRRADIATION PROCESSING OF MATERIALS

(a) Gamma emitters, especially Co^{60}, are used in kilocurie and possibly megacurie quantities for the same purposes referred to previously under "electron-beam processing".

(b) It has been suggested that spent reactor fuel elements be used to obtain the high radiation fluxes required for irradiation processing.

4. STATIC ELIMINATORS

Artificial beta emitters have been suggested for use in radioactive static eliminators, but they are not as efficient for this purpose as the alpha-emitting naturally occurring radionuclides.

5. VOLTAGE PRODUCTION

Several different principles have been utilized in the development of "atomic batteries". These devices may be useful as high-voltage, low-current sources in special instruments.

6. RADIONUCLIDES IN GAUGING

A wide variety of devices for gauging thickness, density, depth, etc., have been developed. Many of these have been adapted to continuous process control. A few examples of such devices are shown in Figure 2.

7. RADIONUCLIDES AS TRACERS

The possible uses of radioactive tracers are unlimited. A few examples of this tremendously useful methodology are shown in Figure 3.

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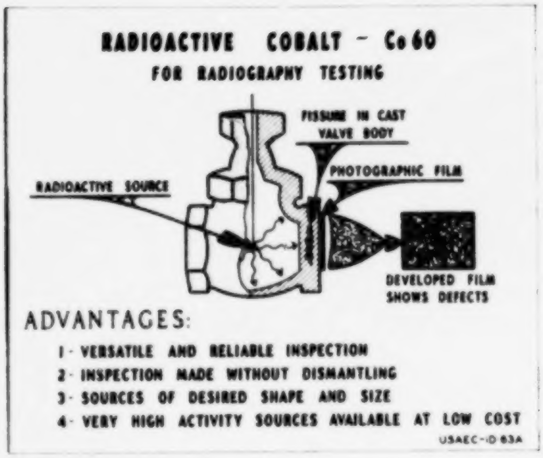
3. McCormick, J. A., "Bibliography on Uses of Radioactive and Stable Isotopes in Industry. A List of Selected References". TID-3511 (July 1957). Technical Information Service Extension, Oak Ridge, Tenn.


RADIOACTIVE COBALT - Co 60

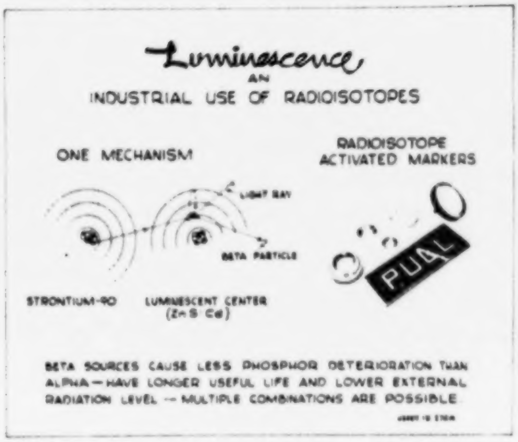
FOR RADIOGRAPHY TESTING

FISSURE IN CAST VALVE BODY
PHOTOGRAPHIC FILM
RADIOACTIVE SOURCE
DEVELOPED FILM SHOWS DEFECTS

ADVANTAGES:

1 - VERSATILE AND RELIABLE INSPECTION
2 - INSPECTION MADE WITHOUT DISMANTLING
3 - SOURCES OF DESIRED SHAPE AND SIZE
4 - VERY HIGH ACTIVITY SOURCES AVAILABLE AT LOW COST

USAEC-ID 63A

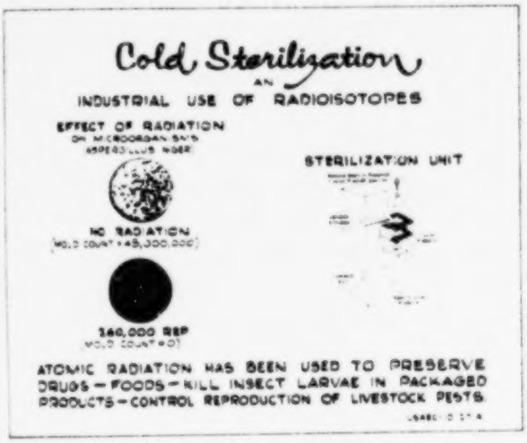

Luminescence

AN

INDUSTRIAL USE OF RADIOISOTOPES

ONE MECHANISM
RADIOISOTOPE ACTIVATED MARKERS
LIGHT RAY
BETA PARTICLE
STRONTIUM-90
LUMINESCENT CENTER (ZnS Cd)
PULL

BETA SOURCES CAUSE LESS PHOSPHOR DETERIORATION THAN ALPHA — HAVE LONGER USEFUL LIFE AND LOWER EXTERNAL RADIATION LEVEL — MULTIPLE COMBINATIONS ARE POSSIBLE

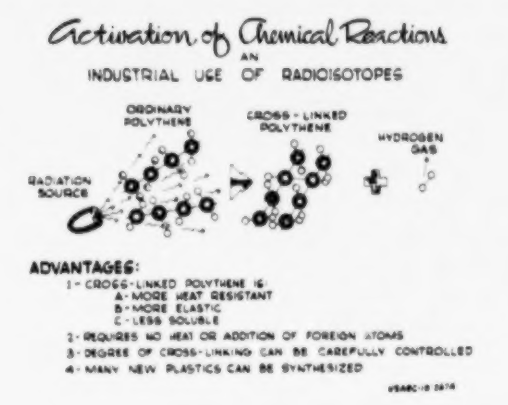

Cold Sterilization

AN

INDUSTRIAL USE OF RADIOISOTOPES

EFFECT OF RADIATION
ON MICROORGANISMS
ASPERGILLUS NIGER
NO RADIATION
(MOLD COUNT 45,000,000)
360,000 REP
(MOLD COUNT 0)
STERILIZATION UNIT

ATOMIC RADIATION HAS BEEN USED TO PRESERVE DRUGS — FOODS — KILL INSECT LARVAE IN PACKAGED PRODUCTS — CONTROL REPRODUCTION OF LIVESTOCK PESTS.


Activation of Chemical Reactions

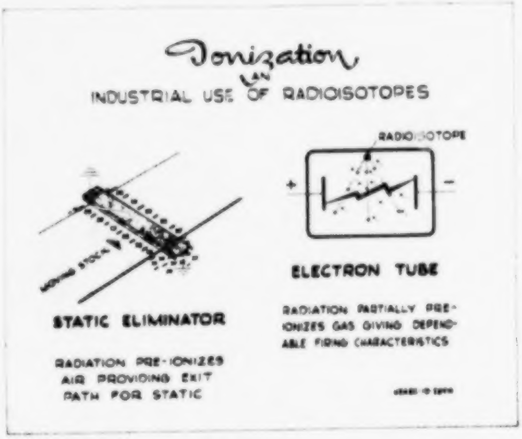
AN

INDUSTRIAL USE OF RADIOISOTOPES

ORDINARY POLYTHENE
CROSS - LINKED POLYTHENE
HYDROGEN GAS
RADIATION SOURCE

ADVANTAGES:

1 - CROSS-LINKED POLYTHENE IS:
    A - MORE HEAT RESISTANT
    B - MORE ELASTIC
    C - LESS SOLUBLE
2 - REQUIRES NO HEAT OR ADDITION OF FOREIGN ATOMS
3 - DEGREE OF CROSS-LINKING CAN BE CAREFULLY CONTROLLED
4 - MANY NEW PLASTICS CAN BE SYNTHESIZED


Ionization

AN

INDUSTRIAL USE OF RADIOISOTOPES

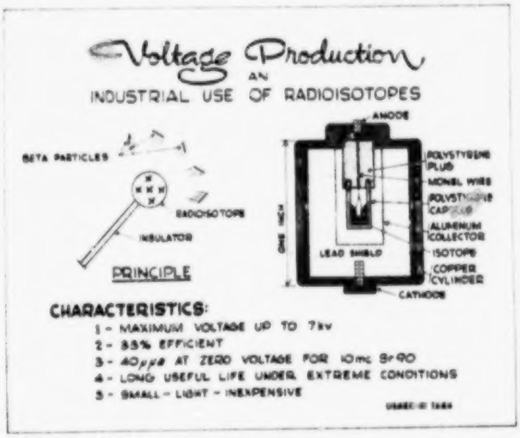
RADIOISOTOPE
+
−
MOVING STOCK

STATIC ELIMINATOR

RADIATION PRE-IONIZES AIR PROVIDING EXIT PATH FOR STATIC

ELECTRON TUBE

RADIATION PARTIALLY PRE-IONIZES GAS GIVING DEPENDABLE FIRING CHARACTERISTICS


Voltage Production

AN

INDUSTRIAL USE OF RADIOISOTOPES

BETA PARTICLES
RADIOISOTOPE
INSULATOR
PRINCIPLE
ANODE
POLYSTYRENE PLUG
MONEL WIRE
POLYSTYRENE CAP
ALUMINUM COLLECTOR
ISOTOPE
COPPER CYLINDER
CATHODE
LEAD SHIELD
ONE INCH

CHARACTERISTICS:

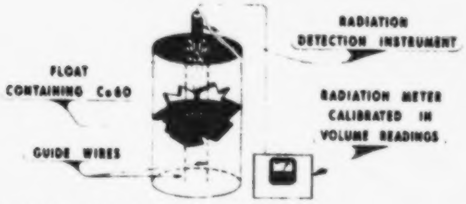
1 - MAXIMUM VOLTAGE UP TO 7kv
2 - 33% EFFICIENT
3 - $40\mu\mu a$ AT ZERO VOLTAGE FOR 10mc Sr90
4 - LONG USEFUL LIFE UNDER EXTREME CONDITIONS
5 - SMALL - LIGHT - INEXPENSIVE

SOME INDUSTRIAL USES OF RADIONUCLIDES

Figure 1

RADIOACTIVE COBALT – Co60

FOR LIQUID LEVEL GAGE


RADIATION DETECTION INSTRUMENT
FLOAT CONTAINING Co60
RADIATION METER CALIBRATED IN VOLUME READINGS
GUIDE WIRES

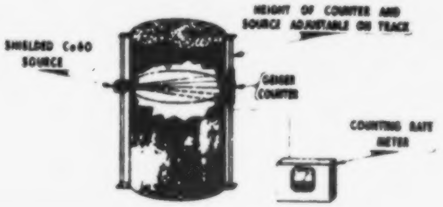
ADVANTAGES:

1- CONTINUOUS RECORDING
2- MEASUREMENT MADE ON CLOSED SYSTEM
3- ADAPTABLE TO AUTOMATIC CONTROL

USAEC-10-180

RADIOACTIVE COBALT – Co60

FOR INDICATING LIQUID HEIGHT


HEIGHT OF COUNTER AND SOURCE ADJUSTABLE ON TRACK
SHIELDED Co60 SOURCE
GEIGER COUNTER
COUNTING RATE METER

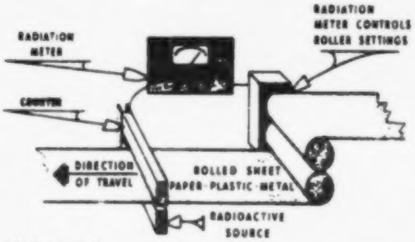
ADVANTAGES:

1- GAGE NOT AFFECTED BY CORROSION AND TEMPERATURE
2- CAN BE OPERATED BY NON-TECHNICAL PERSONNEL
3- ADAPTABLE TO AUTOMATIC RECORDING AND CONTROL OF LIQUID LEVEL

USAEC-10-204

RADIOACTIVE SOURCE

FOR GAGING THICKNESS

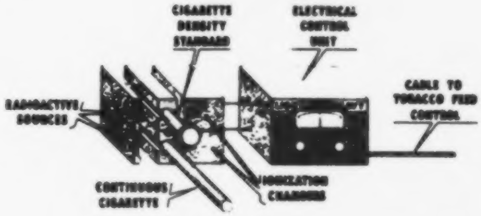

RADIATION METER
RADIATION METER CONTROLS ROLLER SETTINGS
COUNTER
DIRECTION OF TRAVEL
ROLLED SHEET PAPER-PLASTIC-METAL
RADIOACTIVE SOURCE

ADVANTAGES:

1- RADIATION SOURCE SELECTED TO SUIT MATERIAL
2- NO CONTACT - NO TEARING - NO MARKING MATERIAL
3- RAPID AND RELIABLE

GAUGING CIGARETTE FIRMNESS

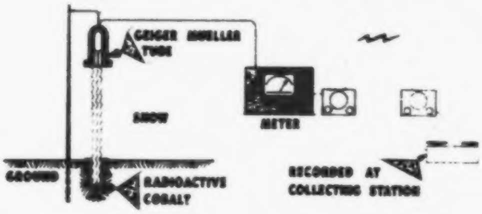
WITH RADIOISOTOPES


CIGARETTE DENSITY STANDARD
ELECTRICAL CONTROL UNIT
RADIOACTIVE SOURCES
CABLE TO TOBACCO FEED CONTROL
CONTINUOUS CIGARETTE
IONIZATION CHAMBERS

ADVANTAGES: 1- MORE UNIFORM FIRMNESS
2- SAVES TOBACCO
3- ELIMINATES MANUAL CONTROL

RADIOACTIVE COBALT – Co60

FOR MEASURING WATER CONTENT OF SNOWFALL


GEIGER MUELLER TUBE
SNOW
METER
GROUND
RADIOACTIVE COBALT
RECORDED AT COLLECTING STATION

ADVANTAGES:

1- MEASURES AMOUNT OF WATER DIRECTLY
2- MEASUREMENTS TAKEN REMOTELY
3- HIGHER DEGREE OF ACCURACY

Detection of Interfaces

IN PIPE-LINE FLOW
(DENSITY GAGE)

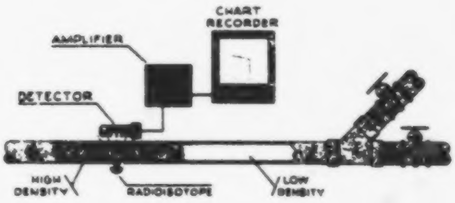
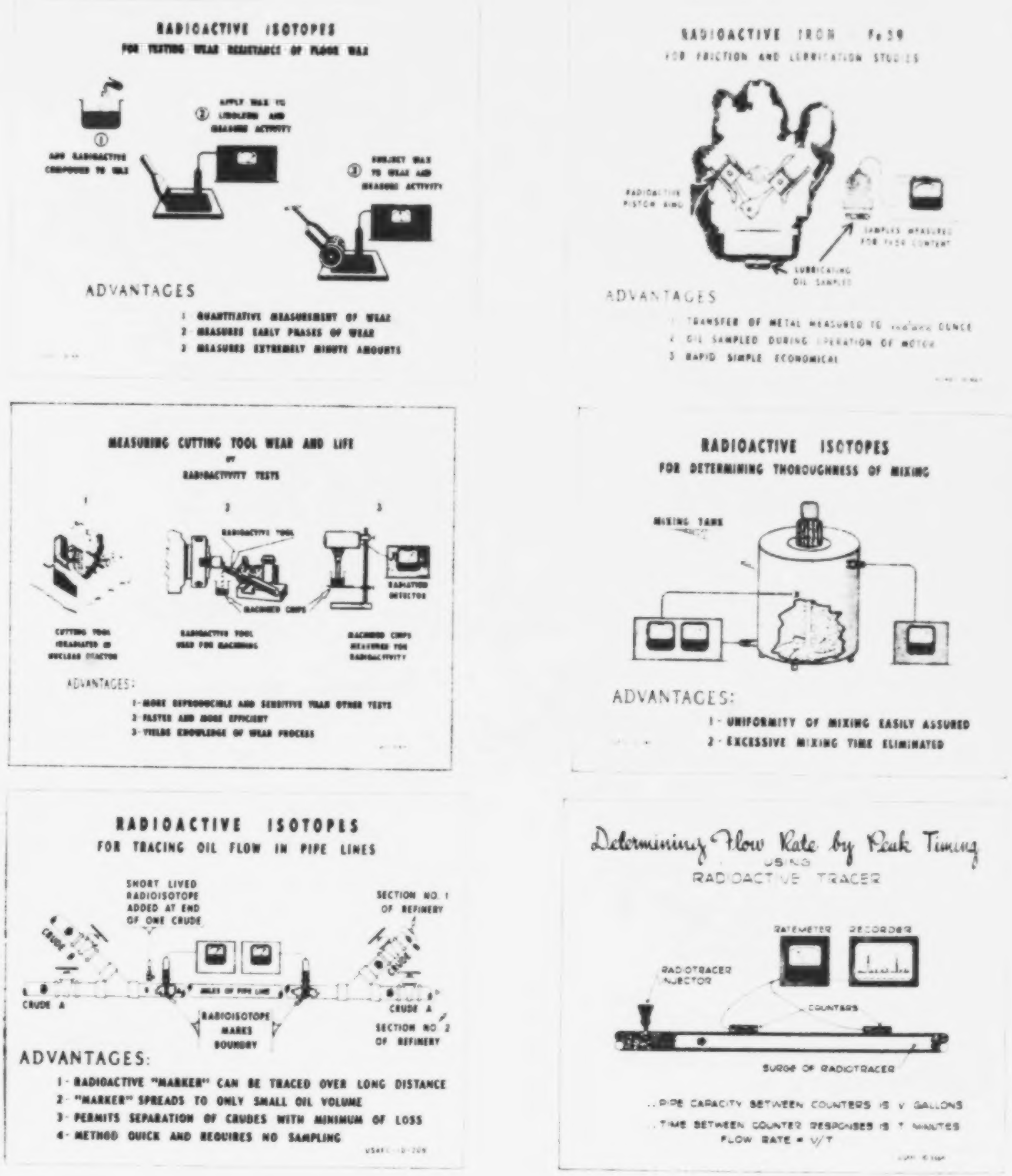


CHART RECORDER
AMPLIFIER
DETECTOR
HIGH DENSITY
RADIOISOTOPE
LOW DENSITY

. PERMITS SEPARATION OF LIQUIDS WITH MINIMUM OF LOSS
. METHOD QUICK AND REQUIRES NO SAMPLING

RADIONUCLIDES IN GAUGING

Figure 2



RADIOACTIVE ISOTOPES
FOR TESTING WEAR RESISTANCE OF FLOOR WAX

① ADD RADIOACTIVE COMPOUND TO WAX
② APPLY WAX TO LINOLEUM AND MEASURE ACTIVITY
③ SUBJECT WAX TO WEAR AND MEASURE ACTIVITY

ADVANTAGES

1 QUANTITATIVE MEASUREMENT OF WEAR
2 MEASURES EARLY PHASES OF WEAR
3 MEASURES EXTREMELY MINUTE AMOUNTS

RADIOACTIVE IRON — Fe 59
FOR FRICTION AND LUBRICATION STUDIES

RADIOACTIVE PISTON RING
SAMPLES MEASURED FOR Fe59 CONTENT
LUBRICATING OIL SAMPLED

ADVANTAGES

1 TRANSFER OF METAL MEASURED TO [illegible] OUNCE
2 OIL SAMPLED DURING OPERATION OF MOTOR
3 RAPID SIMPLE ECONOMICAL

MEASURING CUTTING TOOL WEAR AND LIFE
BY
RADIOACTIVITY TESTS

1
2
3
RADIOACTIVE TOOL
RADIATION DETECTOR
MACHINED CHIPS
CUTTING TOOL IRRADIATED IN NUCLEAR REACTOR
RADIOACTIVE TOOL USED FOR MACHINING
MACHINED CHIPS MEASURED FOR RADIOACTIVITY

ADVANTAGES:

1-MORE REPRODUCIBLE AND SENSITIVE THAN OTHER TESTS
2-FASTER AND MORE EFFICIENT
3-YIELDS KNOWLEDGE OF WEAR PROCESS

RADIOACTIVE ISOTOPES
FOR DETERMINING THOROUGHNESS OF MIXING

MIXING TANK

ADVANTAGES:

1-UNIFORMITY OF MIXING EASILY ASSURED
2-EXCESSIVE MIXING TIME ELIMINATED

RADIOACTIVE ISOTOPES
FOR TRACING OIL FLOW IN PIPE LINES

SHORT LIVED RADIOISOTOPE ADDED AT END OF ONE CRUDE
SECTION NO. 1 OF REFINERY
CRUDE
CRUDE
MILES OF PIPE LINE
CRUDE A
CRUDE A
RADIOISOTOPE MARKS BOUNDRY
SECTION NO 2 OF REFINERY

ADVANTAGES:

1-RADIOACTIVE "MARKER" CAN BE TRACED OVER LONG DISTANCE
2-"MARKER" SPREADS TO ONLY SMALL OIL VOLUME
3-PERMITS SEPARATION OF CRUDES WITH MINIMUM OF LOSS
4-METHOD QUICK AND REQUIRES NO SAMPLING

USAEC-10-209

Determining Flow Rate by Peak Timing
USING
RADIOACTIVE TRACER

RATEMETER
RECORDER
RADIOTRACER INJECTOR
COUNTERS
SURGE OF RADIOTRACER

..PIPE CAPACITY BETWEEN COUNTERS IS V GALLONS
..TIME BETWEEN COUNTER RESPONSES IS T MINUTES
FLOW RATE = V/T

SOME INDUSTRIAL USES OF RADIOACTIVE TRACERS

Figure 3

Sources of Natural Background Radiation Exposure

I. INTRODUCTION

Man has always been exposed to the low-level ionizing radiation present in his environment. This radiation, referred to as "natural background," is not directly caused or produced by man. It is due to natural causes or sources.

Because the exposure which man has received from background radiation causes few discernible effects, a knowledge of the sources of this radiation and the dose man receives from it would help establish levels of exposure which are not expected to have significantly harmful effects on man.

The sources of natural background can be separated into two categories: (1) those found in their original or natural environment, and (2) those which are found not in their natural environment, but have been rearranged or moved by man to an artificial environment.

II. NATURAL SOURCES OF EXPOSURE

Let us look at these two categories of background radiation in more detail.

A. Sources in their Natural Environment

These radiations are from cosmic sources and naturally radioactive elements in the earth's crust.

1. Cosmic Radiation

Primary cosmic rays consist of high energy protons, helium nuclei, and heavy ions. These originate outside the earth's atmosphere, and are present throughout the universe. A small fraction strikes the earth's atmosphere, producing secondary radiations. This gives the earth's inhabitants a continuous radiation dose. These "secondaries" consist of photons, electrons, mesons and neutrons.

Radiation intensity varies with latitude. It is about 10% lower at the geomagnetic equator than at higher latitudes of 45° or greater. More important, however, is the variation of intensity with altitude. The dose at 40,000 ft. is about 100 times greater than at sea level. (See Figure 1.) Cosmic radiation constitutes approximately 25% of the total background dosage to the population.

2. Terrestrial Radiation

Several elements in the earth's crust are radioactive. Those which contribute most

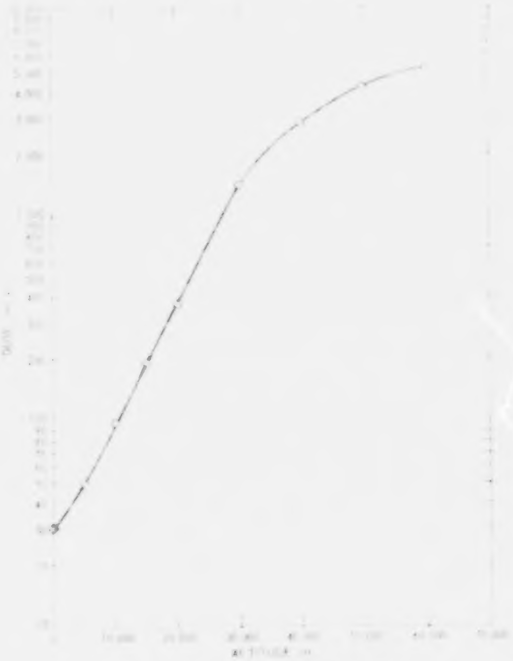


Figure 1.--Cosmic Ray Intensity at Approximately 50° N

to natural background are ^{14}C, ^{40}K, and members of the ^{238}U and ^{232}Th series. On the average, one square mile of soil one foot deep contains about one ton of ^{40}K, three tons of ^{238}U, and six tons of ^{232}Th. Surface rock contains about twice as much uranium and thorium as does the soil. Uranium is found in chemical combination in rock and mineral deposits in Colorado, New Mexico, Utah, and other Western states; and to some extent, in Tennesee and Florida. Thorium appears in large mineral bearing deposits in the Rocky Mountain States and the Carolinas. Potassium-40 is found primarily in granite and alum shales with lesser amounts in soil and sandstone.

Water and air also contain concentrations of ^{14}C, ^{40}K, ^{232}Th and ^{238}U and their daughter products, though to a much lesser extent. The levels of radioactivity in the air vary considerably with location, weather, and atmospheric conditions.

B. Sources in an Artifical Environment

Man disturbs some natural sources of radioactivity by removing them from their natural environment and moving them to an artificial one. By so doing, he tends to concentrate them in some places and diffuse them in others.

1. Uranium Mining and Milling Operations

Because radioactive wastes are discharged into the air and water from mining and milling processes, these operations are considered to add to the background exposure of the population. Here--as in terrestrial radiation--the food, water, and air which man consumes contain concentrations of radiocontaminants. Through there is an external exposure hazard to the mining and milling workers, the dose the general population receives is primarily an internal one.

2. Building Materials

Many of the materials man uses for his buildings contain ^{238}U and ^{232}Th from the earth's crust. The uranium and thorium constitute external sources of exposures. Their daughter products, beginning with the gases radon and thoron respectively, provide internal sources of exposure through the air.

The intensity of radiation which man receives from building materials depends upon the type of structure in which he dwells. The dose rates from wood, brick, and concrete are in ratios of about 1:2:3.

III. OVERALL DOSES FROM NATURAL BACKGROUND

Assuming that a person spends two-thirds of his time indoors and the distribution of buildings in which he spends his time is divided equally, the gonadal dose rate he would receive from external exposure ranges from about 55-105 mrad/yr. This figure includes the dose rate from cosmic and terrestrial radiation as well as from building materials. The total internal gonadal dose rate from mining and milling, terrestrial, and building material sources is about 23 mrad/yr.

A breakdown of doses from the various natural background sources is found in Table 1.

TABLE 1.--Radiation Sources and Doses

Source of Radiation	Dose to Gonads (mrad/yr)
External	
Cosmic rays at sea level	30
Terrestrial plus buildings	25-75
Radon and thoron in air	2
Internal	
Potassium-40	20
Carbon-14	1
Radon, thoron, and daughter products	2
Total dose	80-130

Remember the dose rates listed above are only average figures, giving typical dose rates under "normal" circumstances. The background dose to any person or section of the country can vary from these figures by a factor of two or three.

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Kinsman, S., "Background Radiation Exposures of the General Population," American Industrial Hygiene Association Journal, Vol. 19, No. 1 (Feb. 1958), pp. 8-14.

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Tsivoglou, E.C., et al., Survey of Interstate Polution of the Animas River - Colorado, New Mexico (Cincinnati: U.S. Public Health Service, Robert A. Taft Sanitary Engineering Center, May 1959).

Nuclear Detonations and Fallout

I. INTRODUCTION

Whether detonated in the air, on the earth's surface, under water, or underground (if venting occurs), testing of various nuclear and thermonuclear devices results in the release of materials to the troposphere and occasionally to the stratosphere. (The latter occurs principally as a result of larger explosions.) These radioactive materials can be gaseous or particulate in form and consist of fission products, unfused or unfissioned nuclear materials, and isotopes formed by neutron activation of earth and of debris from structures near the test.

Depending upon the altitude and physical-chemical nature of these particles, they fall back to earth under the influence of gravity and meteorological conditions during a span of seconds to years. When they make contact with the terrestrial and aquatic environment, these materials are more properly called fallout. The terms rainout and snowout may also be used as sub-categories of fallout. This return of radioactive materials to the biosphere via fallout is of health concern because the ultimate deposition of fallout radionuclides in various human "critical organs" may produce long-term biological damage.

II. NUCLEAR AND THERMONUCLEAR DEVICES WHICH CAN CAUSE FALLOUT

Three basic types of devices have been tested for warfare applications: bombs, rocket warheads, and artillery shells. Nuclear explosives used for Plowshare activities are special designs that are being developed for excavation, underground engineering, and other applications.

A. Fission Device (Also called Atomic or Nuclear Weapon)

This device utilizes only fissionable material such as uranium-235 or plutonium-239. The first nuclear device was detonated on a tower at Trinity Site, New Mexico, on July 6, 1945. A blast equivalence of about 20 kilotons of trinitrotoluene (TNT) is common for such a weapon. Both gun-barrel and spherical assemblies have proven successful. The devices may be made small enough to fit into an artillery or bazooka shell.

B. Fission-Fusion Device (Also called Hydrogen or Thermonuclear Weapon)

This device is a combination of a fission device and fusionable material such as deuterium and tritium. The fission releases tremendous quantities of heat which initiate fusion reactions. Fusion events, in turn, release enormous quantities of energy and very high-speed neutrons. The first experimental thermonuclear device was detonated by the United States in the autumn of 1952 at Eniwetok in the Pacific.

III. RADIOACTIVITY PRODUCED AND DISPERSED BY NUCLEAR DETONATIONS

Three types of radioactive material are produced and dispersed by nuclear detonations. These are: unfissioned weapon materials, mixed fission products, and materials with induced activity.

A. Unfissioned Materials

In the earliest designs, detonation efficiencies were low. This means that quantities of unfissioned uranium and plutonium remain after the detonation and can be detected near the site. The extremely toxic radioactive and chemical properties of these materials indicates that their ingestion and inhalation must be avoided. In general, however, unfissioned materials will not appear in significant amounts beyond the distance at which blast and radiation effects are severe.

B. Mixed Fission Products

The activity of the fission products produced in a nuclear explosion represents millions of curies. One minute after the detonation, the gamma activity of the fission products for each kiloton is equivalent to that of 30,000 tons of radium in equilibrium with its daughter products.

Since many of the primary fission products are very short-lived, the radioactivity will decrease rapidly at first. Then the apparent half-life of the mixed fission products will increase with age.

Thus, the dose from this mixture of radionuclides dies away according to the data in Table 1.(1)

Table 1.--Relative Dose Rate from Fission Products*

Time After Detonation	Relative Dose Rate
1 minute	137,000
1 hour	1,000
1 day	20
1 week	2
1 month	.31
1 year	.008
10 years	.00004

*This may be expressed in any unit (i.e., R/hr.) since these are relative values.

Three fission products of biological importance are strontium-90, cesium-137, and iodine-131. It has been estimated that all detonations by the United States, the United Kingdom, France, and U.S.S.R. through 1962 produced approximately 19.3 megacuries of ^{90}Sr, 30 megacuries of ^{137}Cs, and 15 megacuries of ^{131}I.(2)

C. Induced Radioactivity

The species and quantities of radionuclides formed by neutron bombardment depend largely upon the environment of the detonation site. Both short- and long-lived materials are formed, primarily by (n,γ) and (n, p) reactions, and their decay pattern is similar to that for fission products. Typical radionuclides formed in this manner during nuclear detonations are carbon-14, sodium-24, silicon-31, aluminum-28, chlorine-38, cobalt-60, zinc-65, manganese-54, and scandium-46.

It has been estimated that for air bursts 20,000 curies of carbon-14 are produced per megaton of weapon yield. Surface detonations give only about one-half this amount of carbon-14 since not as much atmospheric nitrogen is available for the reaction. The production reaction(3) is as follows:

$$^{14}_{7}N + ^{1}_{0}n \longrightarrow ^{14}_{6}C + ^{1}_{1}H$$

Up until 1963, nuclear weapons tests were responsible for nearly 10 megacuries of atmospheric carbon-14. The amount of natural ^{14}C in the air is estimated as having been four megacuries before the tests, but this does not include the nearly 166 megacuries of natural ^{14}C found on the earth's surface.(4)

During the 1958 U.S. Pacific bomb tests, tungsten was intentionally incorporated into detonation devices so that radioactive tungsten-185 would be produced by a (n,γ) reaction. This radionuclide has provided valuable information on the rate of fallout and has been a tool for studying weather phenomena.(5)

IV. DETONATION FIREBALL AND THE PHYSICAL NATURE OF FALLOUT

After a delay of a few millionths of a second from the time of initial weapon detonation, a luminous gaseous sphere, called a fireball, is formed in the extremely hot atmosphere. The size of the fireball increases with the energy yield of the explosion. The maximum diameter may vary from 900 feet to over 3 miles. Luminosity of the fireball usually does not vary greatly with the energy equivalence of the bomb.

Vaporized bomb materials, water droplets, and debris condense to form the "atomic cloud" as the fireball increases in size and cools. This cloud is red or reddish brown when first formed because of the heat-catalyzed production of nitrogen dioxide, nitrous acid, and other nitrogen compounds from atmospheric gases and water vapor. Eventually the cloud changes to white, due to the water droplets present in it. On or near the surface, a strong updraft called "afterwinds" sucks dirt and debris up from the ground into the radioactive cloud. As the upward thrust of debris slowly subsides, gravity begins to retrieve particulates to the earth's surface. This causes a lengthening and widening of the columnar cloud; eventually, the debris disperses widely due to the jet stream, meteorological conditions, and the continued influence of gravity.

A. Types of Fallout Particles

Fallout particles may be grouped in three categories.

1. Plated-Type Fallout

Vaporized bomb products and induced activities which condense out as a thin shell coating on elevated particles of earth and debris form this type of fallout. Much fallout is of this nature, especially if the detonation takes place at a site relatively devoid of silicate sand.

2. Fused-Type Fallout

Bomb materials may undergo a thermal fusion with silicate to form minute, melted, glassy "beads" which contain air bubbles and mineral grains. These may be transparent to opaque in transmission of light and pale green or yellow to brown or black in color. Radioactivity is distributed irregularly throughout these spherical or irregular forms.

3. Droplet-Type Fallout

If a device is detonated near a body of water, large quantities of water may be vaporized in the fireball. As the vapor condenses out of the cooling atomic cloud, fission products and earth debris may be suspended in water droplets and then fall back to earth as precipitation.

B. Categories of Fallout

Fallout from near-surface bursts may be categorized according to the elapsed time between the time of detonation and complete descent to the earth.

1. First Fallout (also called Local Fallout)

About 80 percent of the radioactive particulates produced by a surface detonation is deposited within a few hours at a distance no further than a few hundred miles from the burst point. Particles which have a diameter of 400 microns, 250 microns, and 150 microns may fall from 80,000 feet in about 1½, 2½, and 5 hours, respectively.[1] It is this first fallout which constitutes the acute radiological hazard. Detonation of a single superweapon may contaminate an area of 5,000 square miles or greater with lethal levels of beta and gamma radiation.

2. Second Fallout (also called Tropospheric Fallout)

Fallout particles of smaller sizes and masses are deposited in the lower atmospheric layers (below 35,000 feet) and are gradually brought down in a matter of one to two months by ordinary weather phenomena. These tropospheric particulates may travel completely around the world in a shell-like fashion, but they usually stay in a rather narrow radioactive band which has the general latitude of the detonation site.

3. Delayed Fallout (also called Stratospheric Fallout)

This type of fallout is created by detonations of weapons of megaton size or greater which propel radioactive particulates of submicron diameters to heights of 16 miles or more. The estimated half-residence time of stratospheric fallout varies from 5 to 60 months. The longer half-residence times result from tropical injections at high altitudes (up to 250,000 feet).

V. DISTRIBUTION OF STRATOSPHERIC FALLOUT

The winds in the stratosphere are predominately easterly or westerly, depending on the time and place. Thus the radioactive cloud which reaches the stratosphere becomes drawn out in a windward direction to form a streak around the globe. The cloud gradually widens in the north-south direction, but changes in altitude are very slow.

Almost the only way for return of this radioactivity from the stratosphere is for the cloud to move into the troposphere where the particles can be brought down by precipitation.

This can happen in the arctic winter by downward movement of the debris through the tropopause, in the arctic spring because of an upward movement through the tropopause, or by turbulent circulation near gaps in the tropopause in each of the temperate zone.

A. Equatorial Testing

Equatorial zone bomb testing (11°N) apparently results in twice as much stratosphere fallout going to the Northern Hemisphere as to the Southern Hemisphere.

B. Large Detonations

In the higher latitudes of the Northern Hemisphere, large detonations apparently will deposit almost all resultant stratospheric fallout in that hemisphere with a mean stratospheric residence time of 1 year or less.

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(4) Report of the Joint Committee on Atomic Energy (86th Congress [Washington, D.C.: U.S. Government Printing Office, Oct. 1960]).

(5) Martell, E. A., and Drevinsky, P. J., "Atmospheric Transport of Artificial Radioactivity," Science, Vol. 132 (Nov. 1960), pp. 1523-1532.

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Calibration of Survey Instruments

Laboratory

I. OBJECT

To calibrate both Geiger-Mueller and ionization chamber type survey instruments and to emphasize the need for calibration before any instrument is used in making radiation measurements.

II. INTRODUCTION

The G-M survey instrument is widely used to measure low intensities of beta and gamma radiation. It is a count-rate instrument in which a meter records the average rate (number/unit time) of beta particles or gamma or X-ray photons that enter the sensitive volume of the G-M tube and produce ionization. The meter reading is dependent only upon the average rate of chamber gas ionization. Due to the radomness of the decay process, the meter needle will fluctuate, particularly on the most sensitive range. The dial ranges usually vary by a factor of ten. We are usually not interested in counting each ionization detected, but rather the intensity or exposure rate in terms of milliroentgens per hour. In order to obtain correct readings of radiation intensity, the survey meter must be calibrated against a standard radioactive source whose activity is known.

To calibrate any range, the meter readings are compared with the corresponding calculated radiation intensities. These radiation intensities can be calculated from the known source activity. The meter should be calibrated over its entire range at a point near each end and at several points in between. When the concern is with X- or gamma radiation, the instrument should be calibrated and used with the probe shield closed. During calibration or use, the instrument should never be brought into a field whose intensity is greater than the maximum scale value or the G-M tube may be damaged. Also, the rate of pulses is so rapid that "jamming" of the instrument results, and the meter needle falls back to zero indicating no radiation present and giving a false sense of security.

The ionization chamber type survey meter is widely used to measure high intensities of X- or gamma radiation. To insure correct measurements, calibration of this type instrument is also necessary.

In the past, radium has been used most frequently for calibration. The radiation exposure rate expressed in mR/hr at various distances from the source can be calculated from:

$$\text{mR/hr} = \text{mg of Ra}/(\text{yards})^2$$

For example, a 100 mg radium source would produce at a distance of one yard, an exposure rate or intensity of:

$$100/1^2 = 100 \text{ mR/hr}$$

At two yards, the exposure rate would be:

$$100/2^2 = 25 \text{ mR/hr}$$

Recently ^{60}Co has been substituted for radium for many purposes, including instrument calibration. The radiation exposure rate at various distances from a ^{60}Co source can be calculated from:

$$\text{mR/hr} = 1.59 \times \text{mCi of } ^{60}\text{Co}/(\text{yards})^2$$

For example, a 100 mCi source of ^{60}Co would deliver an exposure rate at a distance of one yard equal to $1.59 \times 100/1^2 = 159$ mR/hr. At a distance of two yards, the exposure rate would be 1.59 × $100/2^2 = 39.75$ mR/hr. For other exposure rates, use can be made of the inverse square law if the source of radiation is approximately a point source.

III. LIST OF EQUIPMENT

1 G-M survey meter for each group
1 Ion chamber survey meter for each group
1 Small (point) source, 20 to 50 mCi of ^{60}Co

IV. PROCEDURE

Exercise 1: Calibration of G-M Survey Meters

Caution: Be sure to turn off instrument after use. Wear film badge throughout the calibration exercise.

1. Turn on the G-M survey meter. If there is a battery check position, the check should be made. Wait at least 30 seconds or longer for the system to reach stability. Meantime, record survey meter serial number and familiarize self with instrument.

2. Turn range selector switch to the least sensitive scale (highest). If there is no reading, turn range selector switch to the middle range first, and then to the most sensitive range (lowest).

Note if there is any indication of normal background (normal background is usually around 0.02mR/hr or 20 counts/min randomly spaced); the counts may be heard with headphones which can be attached to the instruments.

3. Close beta shield. Turn range selector switch to whatever setting results in a one-fourth to one-half scale deflection at the calibration line most distant from the source.

4. Note the average reading and the distance from the source. Source-to-instrument distances should be measured to the center of the probe held perpendicularly to the radiation flux.

5. Move toward the source, taking a number of meter readings sufficient to accurately calibrate the scale. Scale reading, range and distance must be noted and recorded on the data sheet.

6. Repeat the above procedures for any other range that may be designated for calibration.

7. Turn the instrument control dial to "OFF" position.

8. From the distance values noted, calculate the exposure rates present (mR/hr) at these distances for which meter readings were obtained. (For ease of calculation, use distances that are either yards or feet, using smaller intervals near the source and larger ones as you move farther away. It is sufficient for our purposes to obtain 4 or 5 points, distributed over the entire range. This calculation may be done first.)

9. Upon completion of the calculations for all points at which readings were taken, the calibration curve for each range calibrated should be constructed on linear graph paper. To plot calibration curves, dial readings should be plotted on the ordinate (Y axis) and calculated exposure rates on the abscissa (X axis). Report the instrument serial number, name of manufacturer, date, type and strength of the radioactive source used for calibration, and the person(s) doing the calibration on the data sheet.

Exercise 2: Calibration of Ionization Chamber

1. Close the beta shield, turn the instrument on letting it warm up for several minutes, and adjust to zero. Unless this instrument is thoroughly warmed up, the zero setting will drift.

2. After the instrument is zeroed, turn the range selector switch to the lowest, most sensitive range. Repeat procedure as outlined for G-M instruments.

V. QUESTIONS

1. Which of the two types of survey instruments used is the most sensitive to low intensities of radiation?

2. What two kinds of radiation will these instruments detect in general? Which of these two kinds of radiation is measured more accurately in mR/hr by these instruments?

3. In general, when the G-M survey meter reading decreases, the radiation level is less. However, name two other conditions which would cause this same effect.

4. What would be the effect upon your readings while calibrating these instruments if you moved from one distance to another too rapidly?

Data Sheet

Distance (Feet or Yards)	Range Selector Setting	Dial Reading (mR/hr)	True mR/hr (Calculated)

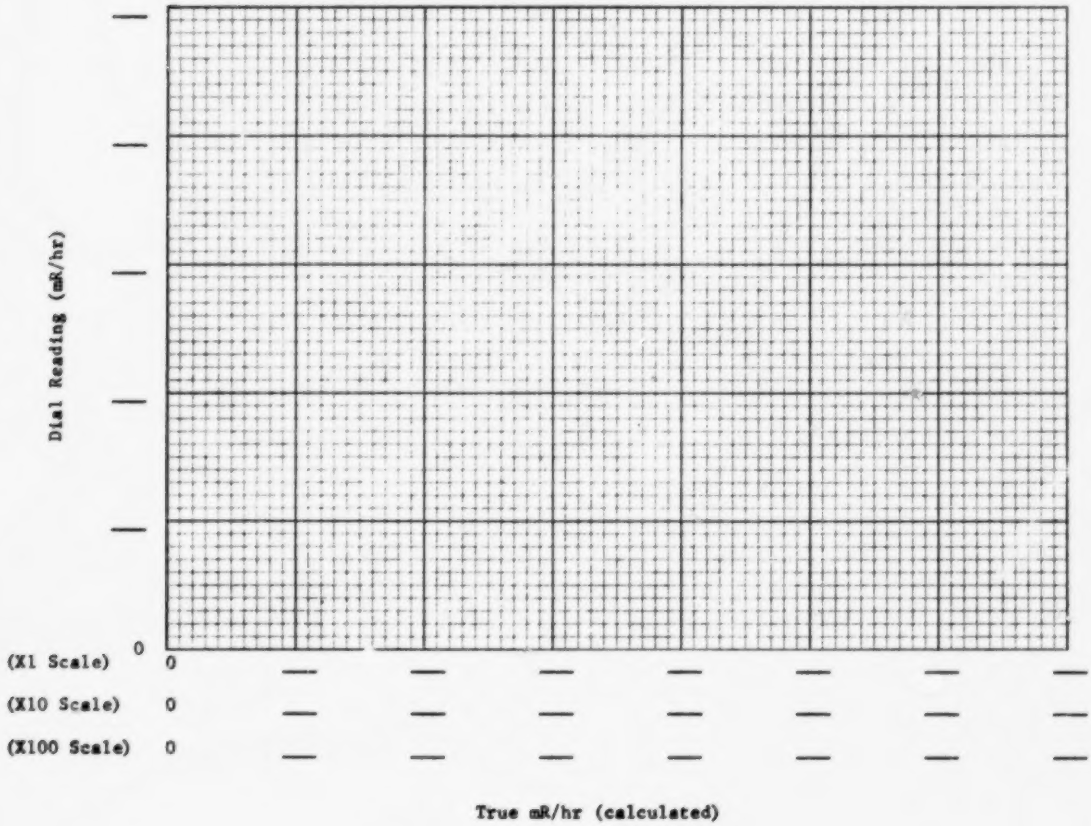
CALIBRATION CURVE

Current Date ______________________

Source __________________________ Date of Calibration ________________

Type of Instrument ________________ Serial No. ______________________

Manufacturer _____________________ Model No. ______________________



Data Sheet

Distance (Feet or Yards)	Range Selector Setting	Dial Reading (mR/hr)	True mR/hr (Calculated)

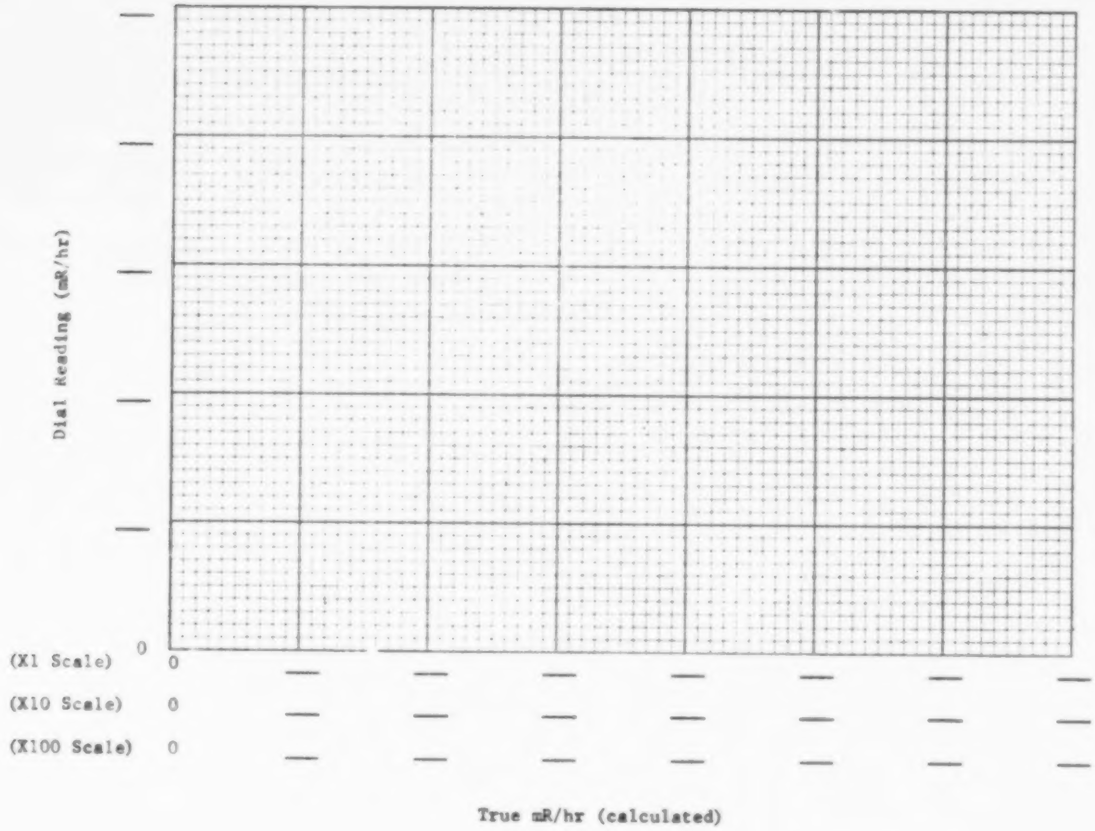
CALIBRATION CURVE

Current Date ______________________

Source ______________________ Date of Calibration ______________________

Type of Instrument ______________________ Serial No. ______________________

Manufacturer ______________________ Model No. ______________________



Introduction to Laboratory Counting Techniques

OBJECT:

1. To familiarize course participants with the operation of laboratory counting instruments.

2. To determine the characteristic response curves for GM and Internal Proportional Counters.

3. To determine representative background counts.

INTRODUCTION:

A laboratory counting instrument records the number of ionizing events that occur in the detecting element. These counts, when taken over proper time periods, and when corrected by proper factors or by comparison with adequate standards, are a measure of the activity present in a radioactive sample.

The purpose of the laboratory counter-scaler is to determine, more accurately than is possible with a survey meter, the radioactivity of samples in terms of millicuries, microcuries, or similar units. This is known as absolute counting. It may also be desired to know simply the relative number of counts occurring in the material before and after a given process, in which case no reference is required to an absolute disintegration rate. The comparison may then be made in counts per minutes or counts per second per unit volume or mass of sample.

Before samples are counted, certain operating characteristics of the detector and circuit should be known. A characteristic plateau curve for the detecting element should be determined.

The characteristic curve indicates the variation in counting rate with change of applied voltage, all other parameters being held constant. Laboratory line voltages change as electrical loads are varied in other parts of the building, and this produces a corresponding change in the high voltage applied to the detector. It is desirable to operate the detector in a voltage region such that line voltage variation will have a minimal effect on the count rate.

The sample shelf structure within the lead shield for the GM tube has five shelf positions. The second shelf from the top is generally used as the standard position. However, if the sample to be measured produces too high a count rate when in the second shelf position, it can be placed on a lower shelf where a lower efficiency and consequently a lower count is found. If absolute disintegration rates are desired and the sample is moved from one shelf to another, appropriate geometrical efficiency corrections must be made.

Usually, the weakest sample that may be counted conveniently is one that gives a count of twice background. The lead shield enclosing the GM tube and sample mount serves to reduce the background count, allowing for greater accuracy in the measurement of weak sources. This lead shield, or "pig", also permits counting by several instruments in close proximity without interference with each other. Background counts are obtained by operating the instrument with an empty sample holder in place.

The importance of geometry and sample mounting must be emphasized. The sample should always be mounted firmly on the same material, preferably light materials such as lucite, cardboard, or aluminum; and an arrow, circle, or some other identifying mark should be drawn on the sample holder to aid in reproducing the position. Reproducibility of counting results within statistically acceptable limits is largely dependent upon precise positioning of the sample.

LIST OF EQUIPMENT:

End Window GM Counter-Scaler

Gas Flow Internal Proportional Counter-Scaler

UX_2 Beta-standards

Ra D and E standards

PROCEDURE

1. STARTING SCALER

The instructor will describe the counting equipment and demonstrate the proper procedure for positioning the sample and placing the counting equipment in operation. Brief instructions on the operation of the counter are given as follows:

a. Place the sample in the sample mount.

b. See that the "High Voltage", and "Count" switches are off.

c. Turn "High Voltage" control(s) to their minimum voltage position.

d. If scaler has been allowed to warm up for two minutes or more, turn on "High Voltage" switch, press the "reset" buttons.

e. Turn on "Count" switch and adjust "High Voltage" control to give specific voltage for proper tube operation. (Voltage should be specified on label attached to counter.)

2. DETERMINATION OF GM TUBE CHARACTERISTIC CURVE

a. Beginning with the lowest voltage available, increase the voltage until counts begin to be recorded by the scaler. Record this "starting voltage".

b. Round off the starting voltage to the nearest 50 volts and take a 1-minute count at this voltage setting. Record volts and c/m obtained.

c. Increase the voltage by 50-volt steps and take 1-minute counts at each step.

d. Plot the data from each step as a point on linear graph paper ruled with "voltage" as abscissa (x-axis) and "c/m" as ordinate (y-axis). PLOT EACH POINT BEFORE GOING TO THE NEXT 50 VOLT STEP!

e. Stop and lower the voltage if the curve commences to rise sharply. In any case, stop when a voltage of 1900 is reached, as GM tubes are occasionally observed to fail very abruptly at higher voltages.

f. Add to the characteristic curve the number affixed to the shield holding the GM tube, and mark with a line of red pencil upon the "High Voltage" meter glass the proper operating voltage for the GM tube.

3 DETERMINATION OF THE INTERNAL PROPORTIONAL COUNTER CHARACTERISTIC CURVE

The Internal Proportional Counter is characterized by two plateaus - one on which the ionization produced by alpha particles only is sufficient to activate the scaler, (beta particles are not counted), and another plateau on which the scaler counts pulses from both the alphas and the betas.

a. Place the alpha- and beta-emitting sample uncovered in the chamber, and flush out the air with at least a 100 second gas purge.

b. Set the timer to one minute, and its switch to the count position.

c. Raise the voltage slowly until the scaler begins to record counts. Record this "starting voltage".

d. Round off the starting voltage to the nearest 50 volts and take a 1-minute count at this voltage setting. Record volts and c/m obtained.

e. Increase the voltage by 50 volt steps and take 1-minute counts at each step.

f. Plot the data from each step as a point on linear graph paper ruled with "voltage" as abscissa(x-axis) and "c/m" as ordinate (y-axis). Plot each point before going to the next step. The end of the alpha plateau is indicated by a sharp rise in the c/m.

g. Continue counting and plotting at 50 volt intervals until the end of the beta plateau is reached, as indicated by another rise in the count rate. Unlike the GM tube, the internal proportional counter will not be damaged by excessive voltage.

h. Add to the characteristic curve the instrument number, and indicate the proper operating voltage for each plateau. For internal proportional counters the operating voltage is usually chosen as the center of the respective plateau.

4. THE "BACKGROUND" COUNT OF THE INSTRUMENT SHOULD BE MADE.

a. Remove sample from counter.

b. Count in the normal manner for as many minutes as time permits or the instructor directs.

c. Record background as c/m.

This outline was prepared by C. D. Geilker, Health Physicist, Radiological Health Training Activities, Training Program, SEC.

INTRODUCTION TO COUNTING TECHNIQUES DATA SHEET

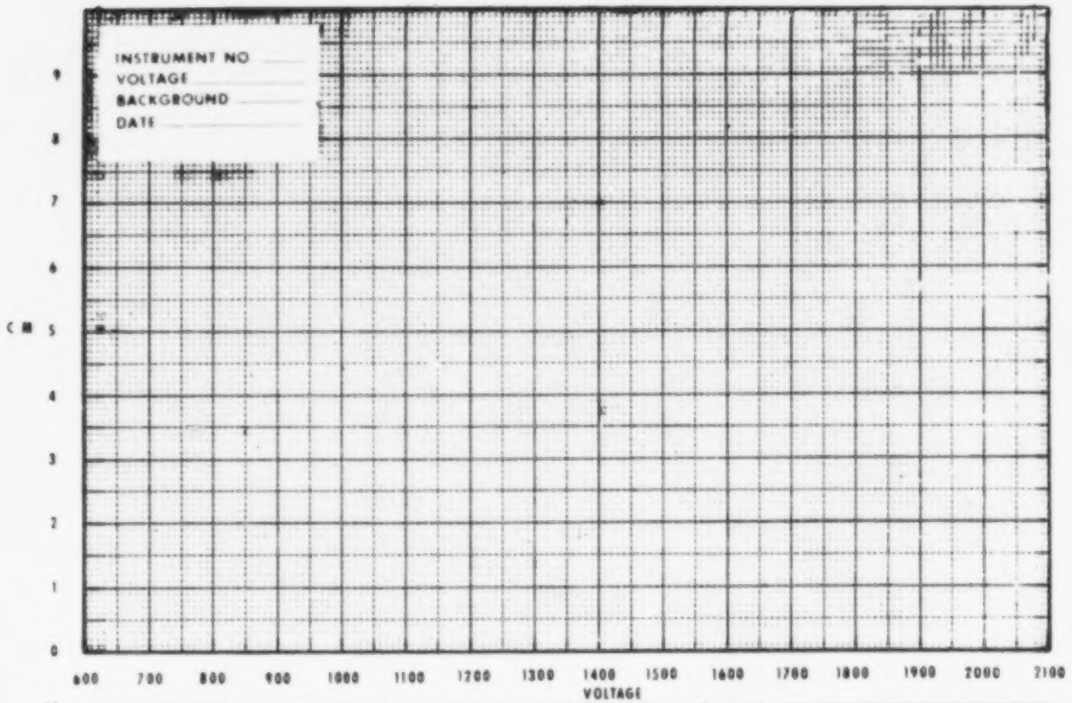
Name______________________________ Date__________

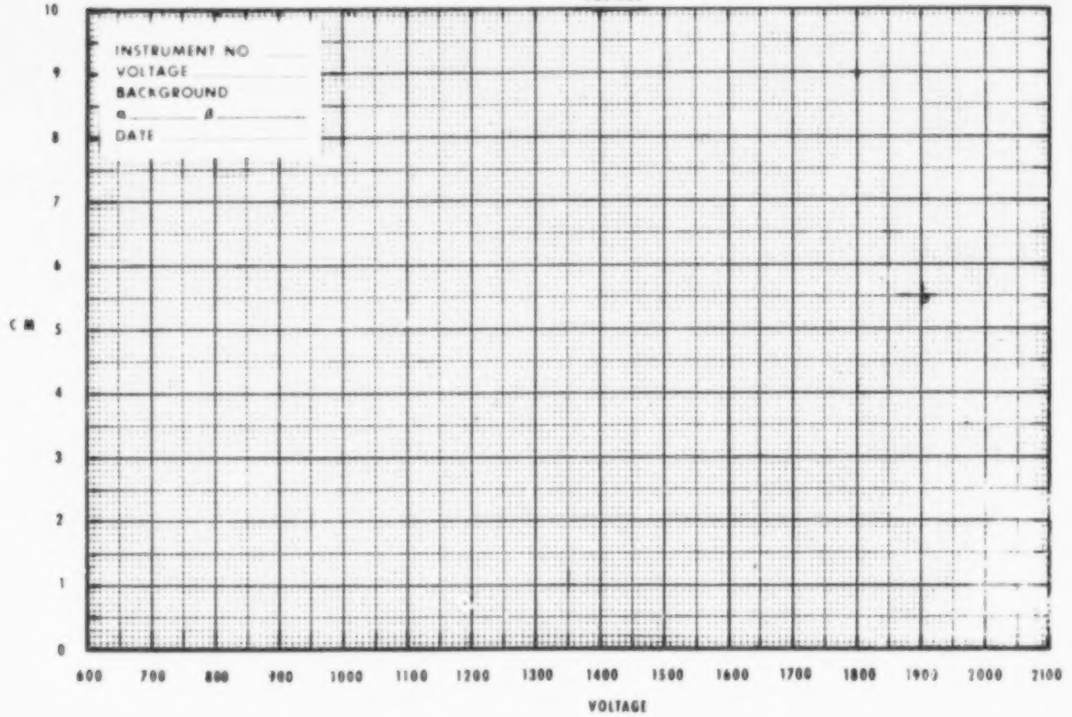
GM COUNTER SERIAL NO. ________

VOLTAGE	c/m	VOLTAGE	c/m
850		1400	
900		1450	
950		1500	
1000		1550	
1050		1600	
1100		1650	
1150		1700	
1200		1750	
1250		1800	
1300		1850	
1350		1900	

INTERNAL PROPORTIONAL COUNTER SERIAL NO. ________

VOLTAGE	c/m	VOLTAGE	c/m
600		1400	
650		1450	
700		1500	
750		1550	
800		1600	
850		1650	
900		1700	
950		1750	
1000		1800	
1050		1850	
1100		1900	
1150		1950	
1200		2000	
1250		2050	
1300		2100	
1350			


INSTRUMENT NO
VOLTAGE
BACKGROUND
DATE
C M
0
1
2
3
4
5
6
7
8
9
600
700
800
900
1000
1100
1200
1300
1400
1500
1600
1700
1800
1900
2000
2100
VOLTAGE


INSTRUMENT NO
VOLTAGE
BACKGROUND
α
β
DATE
C M
0
1
2
3
4
5
6
7
8
9
10
600
700
800
900
1000
1100
1200
1300
1400
1500
1600
1700
1800
1900
2000
2100
VOLTAGE

GPO 876-280

INTRODUCTION TO COUNTING TECHNIQUES DATA SHEET

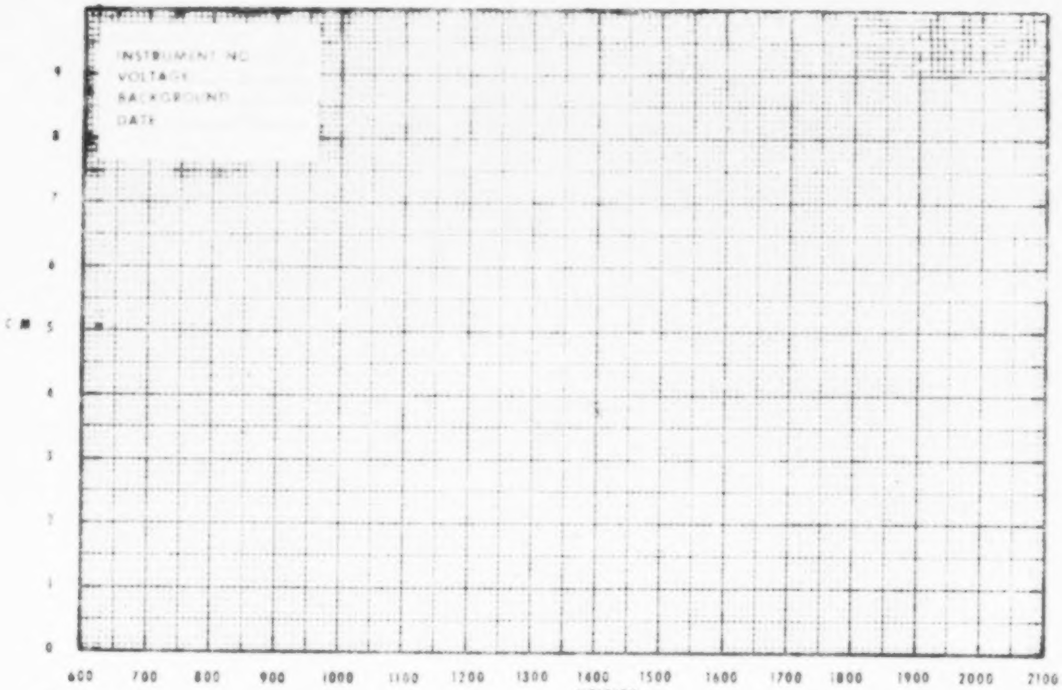
Name ____________________ Date __________

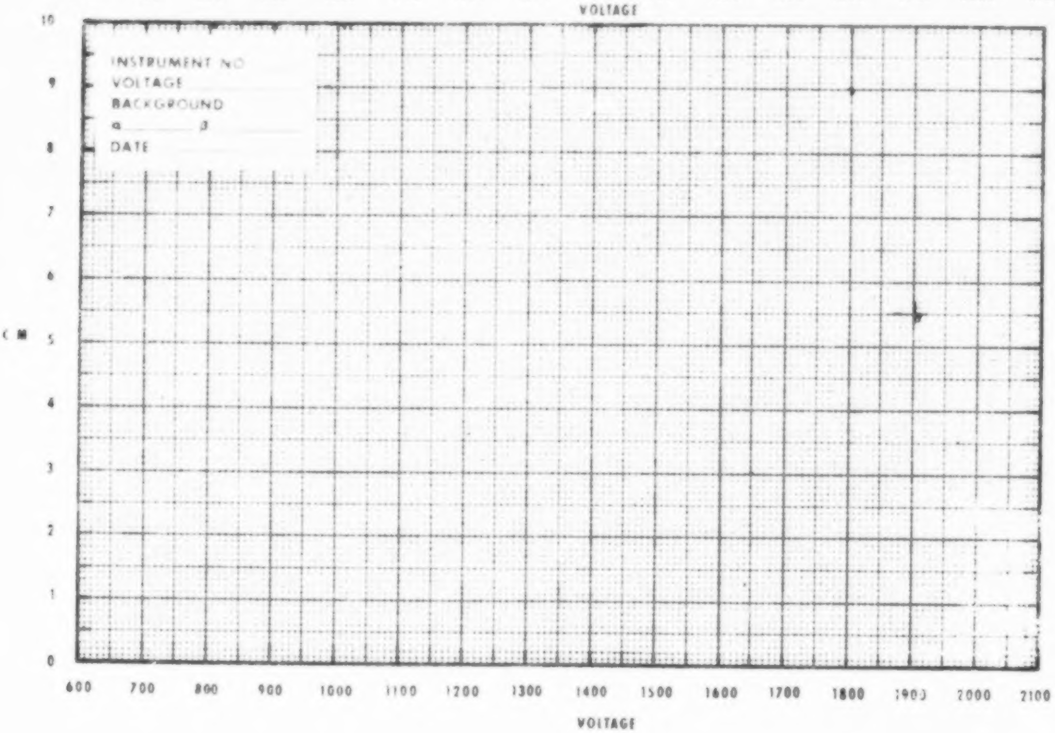
GM COUNTER SERIAL NO. __________

VOLTAGE	c/m	VOLTAGE	c/m
850		1400	
900		1450	
950		1500	
1000		1550	
1050		1600	
1100		1650	
1150		1700	
1200		1750	
1250		1800	
1300		1850	
1350		1900	

INTERNAL PROPORTIONAL COUNTER SERIAL NO. __________

VOLTAGE	c/m	VOLTAGE	c/m
600		1400	
650		1450	
700		1500	
750		1550	
800		1600	
850		1650	
900		1700	
950		1750	
1000		1800	
1050		1850	
1100		1900	
1150		1950	
1200		2000	
1250		2050	
1300		2100	
1350			


INSTRUMENT NO
VOLTAGE
BACKGROUND
DATE
CM
0
1
2
3
4
5
6
7
8
9
600
700
800
900
1000
1100
1200
1300
1400
1500
1600
1700
1800
1900
2000
2100
VOLTAGE


INSTRUMENT NO
VOLTAGE
BACKGROUND
α
β
DATE
CM
0
1
2
3
4
5
6
7
8
9
10
600
700
800
900
1000
1100
1200
1300
1400
1500
1600
1700
1800
1900
2000
2100
VOLTAGE

Determination of Half-Life

OBJECT

To acquaint course participants with a laboratory method of determining the half-life of a radionuclide.

INTRODUCTION:

Determination of the half-life of a radionuclide may be readily accomplished in the laboratory by the graphical method of plotting activity of the sample (c/m) vs. elapsed time. The half-life is then that time interval required for the activity to decrease by one-half and the shape of the curve reveals information about the purity of the sample.

LIST OF EQUIPMENT

Short half-life sample
Laboratory counters

PROCEDURE:

(1) Place the sample in the counter, make certain the counter is properly set up (voltage, gas flow, shelf, etc.), and count for one minute.

(2) Record c/m on Counting Data Sheet.

(3) Repeat count at five minute intervals.

(4) Mark the linear (horizontal) axis of the graph sheet in time units, beginning not with the time of first count, but with the hour of preparation of the sample (obtain from instructor).

(5) Plot c/m on the log (vertical) axis of the graph sheet, at the correct "clock time of count" on the horizontal axis.

(6) After eight or more points have been plotted, connect with a straight line, and determine the half-life.

QUESTIONS:

(1) What is the apparent half-life?

(2) Is the sample a single radionuclide, (i.e., does the line continue straight upon plotting further points)?

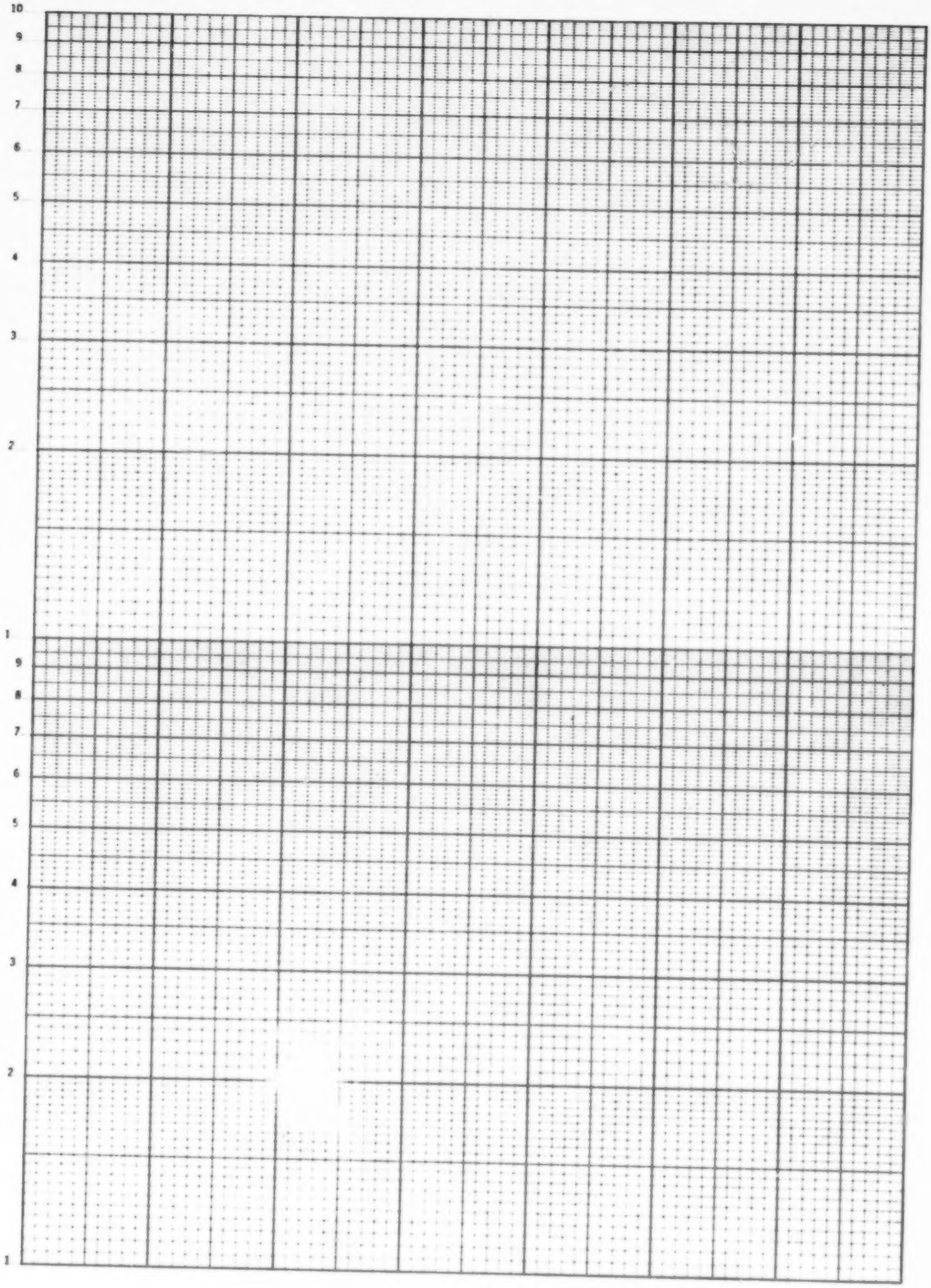
Laboratory Counting Data Sheet

Operator's Name ______________________ Date ____________________

Counter Type _______________ Serial No.: Counter ___________ Shield ___

Operating Voltage ________________________

Sample	Time of Count	Total Count	Gross c/m	Background	Net c/m	


10
9
8
7
6
5
4
3
2
1
9
8
7
6
5
4
3
2
1

Efficiency of Counting

OBJECT:

To indicate the effect of several factors on the efficiency of counting.

INTRODUCTION:

There are several factors which affect the efficiency of any counting arrangement. These include geometry, absorption, scattering, and resolving time.

Geometry is that percent of solid angle about the source which is subtended by the face of the detecting volume. The relative physical positions of the source and the detector establish this factor; the closer the two are, the larger the geometry factor, and the greater the percentage of the total emissions detected.

Scattering is the second most significant factor affecting counting efficiencies. The most important scattering action is that called "back-scatter" whereby radiations originally directed away from the detector are "reflected" back toward the sensitive volume by the source support material. This effect increases with the energy of the radiation, the atomic number of the support material, and its thickness. Increases of as great as 50% can be effected by back-scatter. A second significant scattering action is involved with the use of absorbers to determine the energies of beta and gamma radiation. Absorbers will scatter radiation into the tube thus decreasing the apparent absorption. Errors of as great as 25% can be introduced by this action. Other scattering is produced by the wall of the shield, the mass of the sample itself, and the air between the source and detector.

Absorption is a third factor affecting counting and is significant in special cases. Absorption of radiation is generally accomplished in three locations; the mass of the sample itself; the air between the sample and the detector; and the window of the detector. This absorption is generally significant only when the radiation involved is alpha or weak beta, and/or the sample is relatively thick, as in the case of dried food or sewage samples, or liquid samples.

The above factors become significant if results are to be expressed in absolute units (micro-curies) or if two assays are to be compared. Either corrections must be made to compensate for differing effects, or conditions must be identical in both samples to eliminate introduction of differential errors.

List of Equipment:

One Laboratory G-M Instrument for each group of trainees

One reference source beta emitter for each group of trainees

One isotropic source for each group of trainees

One set of different source backing materials for each group of trainees

Procedure:

G-M Counter No.____________________Operating Voltage ____________________

Date ____________________ Background ____________________cpm

A. Determine over-all counting efficiency

Using a reference source, determine the counting efficiency for the five shelf positions of the G-M counter by determining the cpm/dpm ratio for each shelf.

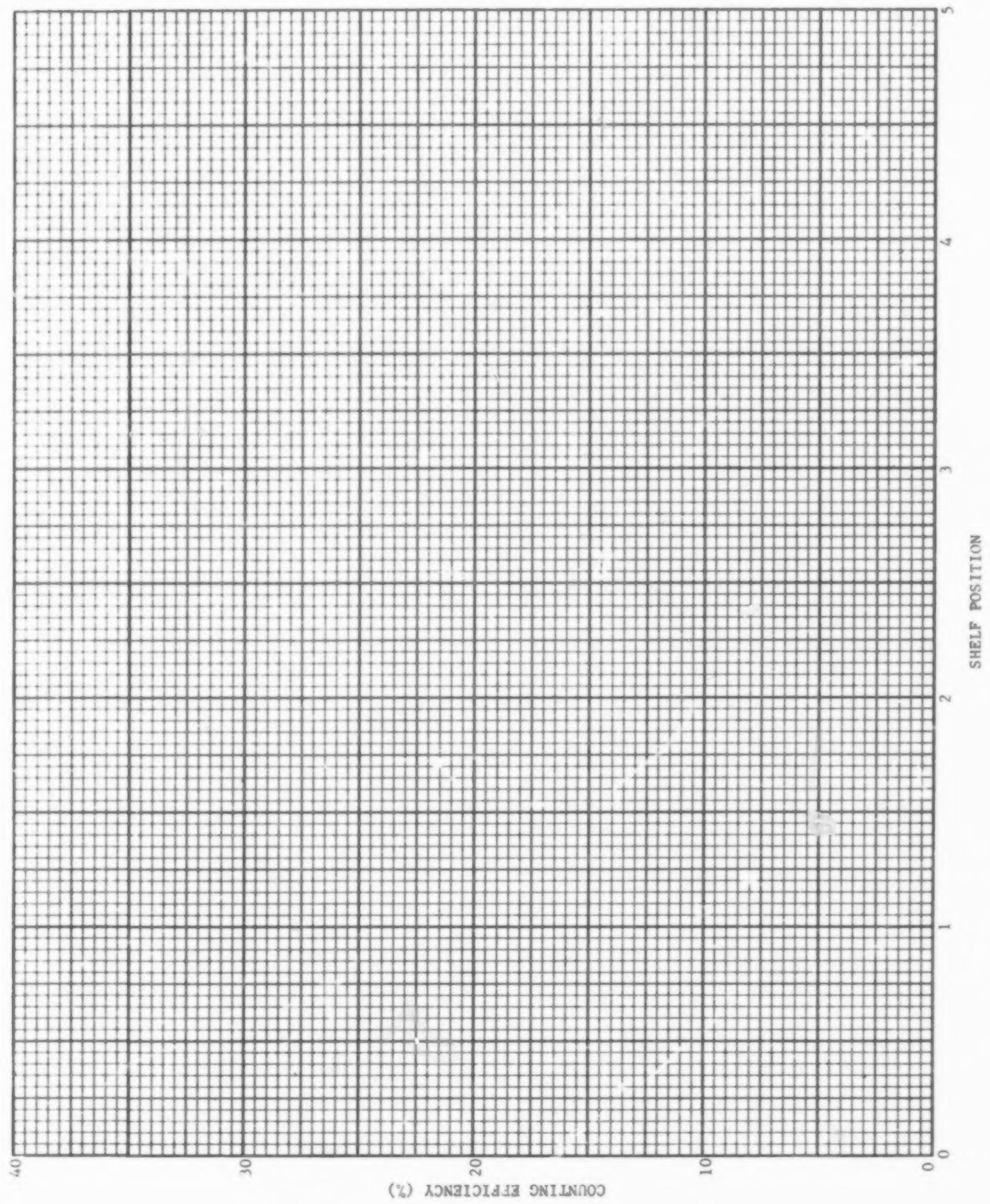
Reference Source Material ____________________ ; Reference Source No. ____________________

Calibrated Activity____________________dpm; Date of Calibration ____________________

Elapsed time between calibration date
and date of experiment____________________ ; No. of Half-lives ____________________

Fraction Remaining ____________________ Present Activity ____________________

Shelf No.	Gross Counts Per Min.	Background Counts Per Min.	Net Counts Per Min.	Counting Efficiency (cpm/dpm)
1				
2				
3				
4				
5				


SHELF POSITION
0
1
2
3
4
5
COUNTING EFFICIENCY (%)
0
10
20
30
40

B. Backscattering vs. Atomic Number

Using the isotropic source, determine the effect of various source support materials on the counting rate observed.

1. Count source on the shelf indicated by the instructor.

2. Count source with backing materials provided and compute backscatter factors (B.F.) where B.F. = cpm with backing/cpm without backing.

3. Repeat the procedure using a source with a different energy or the same source on a different shelf.

SOURCE NO. 1

Type of Radiation

Source Material

Shelf No.

Energy

Backing Material	Gross c/m	Back-ground	Net c/m	B.F.
No Backing				
Aluminum				
Steel				
Copper				
Cadmium				
Silver				
Lead				

SOURCE NO. 2

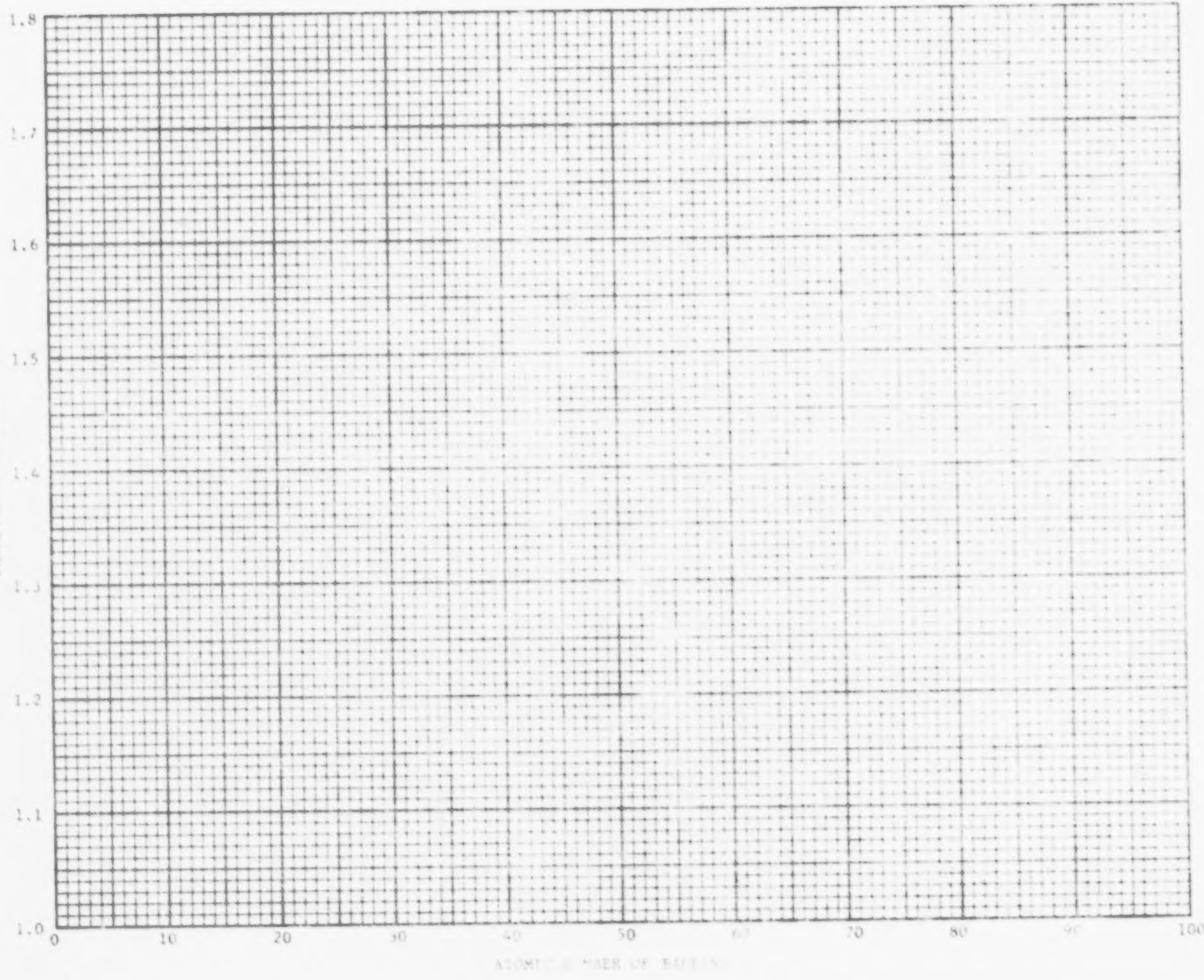
Type of Radiation

Source Material

Shelf No.

Energy

Backing Material	Gross c/m	Back-ground	Net c/m	B.F.
No Backing				
Aluminum				
Steel				
Copper				
Cadmium				
Silver				
Lead				


1.8
1.7
1.6
1.5
1.4
1.3
1.2
1.1
1.0
BACKSCATTER FACTOR
0
10
20
30
40
50
60
70
80
90
100
ATOMIC NUMBER OF BACKING

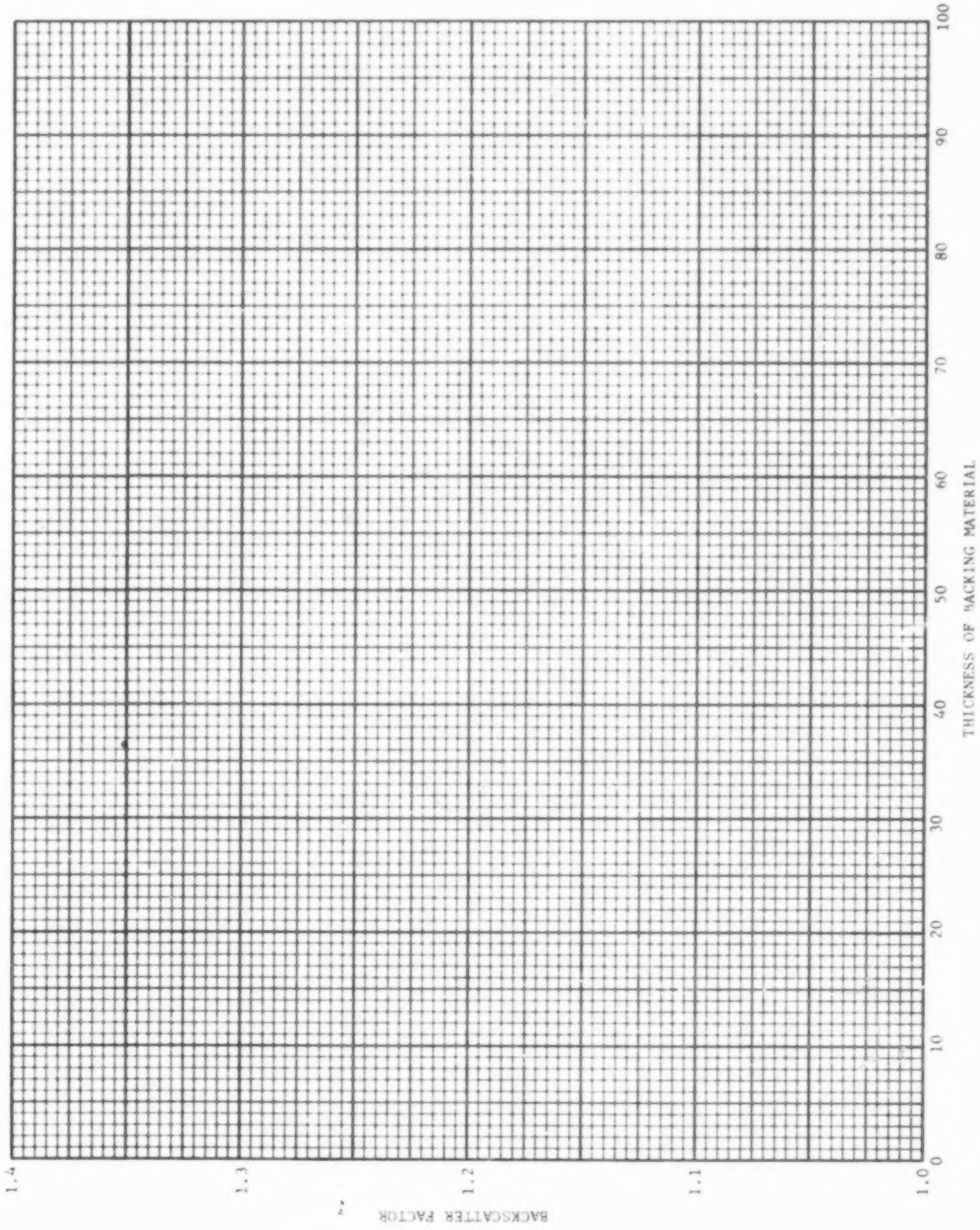
C. Backscattering as a function of Thickness

Using the isotropic source, determine the effect of increasing thickness of backing material using thin aluminum foils.

Backing Material ______________ Type of Radiation ______________

Shelf No. ______________ Energy of Radiation ______________

Thickness of Backing mg/cm^2	Gross Counts Per Min.	Background Counts Per Min.	Net Counts Per Min.	Backscatter Factor (cpm with scatterer/ cpm with no backing)


THICKNESS OF BACKING MATERIAL
0
10
20
30
40
50
60
70
80
90
100
BACKSCATTER FACTOR
1.0
1.1
1.2
1.3
1.4

D. Determine GM tube resolving time

1. Place both halves of the split sample next to each other in the sample holder. Then place the sample holder on a shelf to cause about 20,000 cpm to be detected.

2. Count the sample for at least two minutes and record the count rate as R_{12}.

3. Carefully remove one of the split samples without disturbing the position of the other sample and count the remaining split sample for at least two minutes. Record the count rate as R_1.

4. Carefully replace the split sample that was removed in step 3 in exactly the same position that it was before it was removed. Now remove the other split sample.

5. Count the remaining split sample for at least two minutes. Record count rate as R_2.

6. Calculate the detector resolving time by using the formula:

$$T = \frac{R_1 + R_2 - R_{12}}{2\,(R_1 \times R_2)}$$

Where T = resolving time (seconds)
R_1 = count rate of source 1 (counts per second)
R_2 = count rate of source 2 (counts per second)
R_{12} = count rate of combined sources 1 and 2 (counts per second)

Sample	Time of Count	Total Count	Gross c/sec	Background	Net c/sec
R_{12}					
R_1					
R_2					

The true sample count rate is related to the resolving time by the following formula:

$$R = \frac{R_o}{1 - R_o T}$$

Where R = true counting rate (c/s)
R_o = observed counting rate (c/s)
T = resolving time, seconds

Introductory X-ray Laboratory

OBJECT

To study the parameters affecting x-ray intensity and quality, to observe some of the properties of x-rays, and to demonstrate the reduction of patient dose in diagnostic radiology.

INTRODUCTION

Control of x-ray intensity and quality is accomplished by three variables, tube current, kilovoltage, and filtration. Tube current is measured in milliamperes (ma). Applying more power to the filament increases the filament temperature, which in turn increases the flow of electrons from filament to anode. It should not be confused with filament current. Kilovoltage determines the velocity of the electrons and thus the energy released on impact with the target. Filtration selectively absorbs the less penetrating radiation, and produces a beam which is more nearly monoenergetic.

The focal spot, where the electrons strike the target, is sufficiently small that it can be considered a point source. Therefore the inverse square law will apply in the relation of distance to dose rate.

Reduction of patient dose can be accomplished by limiting the volume of tissue irradiated, and by selecting the operating conditions so as to minimize the entrance dose without impairing the quality of the picture.

EQUIPMENT

Personnel monitoring instruments for each trainee;

X-ray unit, with cones and aluminum filters

Roentgen rate meter, with high and low range probes

Yardstick

Ring stand and clamp

Portable protective barrier

Phantom

Ionization chamber survey meters

CAUTION

In all the six parts of this exercise, the x-ray beam will be directed horizontally away from the class. It is important that everyone stay out of the cone of radiation at all times. The x-ray unit will be equipped with the long cone except when the effects of no cone and short cone are to be demonstrated.

PROCEDURE

1. Effect of the tube current on dose rate

Support high range probe of roentgen ratemeter on ring stand 36" from the tube target. Turn selector switch to the 10 r/min scale and adjust zero. Set x-ray unit to operate at 75 kvp with no filtration, or under such other conditions as the instructor may direct. Keeping all factors constant except ma, record dose rate at 1 ma intervals from 1 to 7 ma.

ma	r/min
1	
2	
3	
4	
5	
6	
7	

Graph data on opposite page. In this and subsequent parts of the exercise, repeat any reading which seems to be out of line.

2. Effect of kilovoltage on dose rate

These data will be taken at 36", operating the x-ray unit at 4 ma with no filtration, or under other conditions as the instructor may direct. Holding all factors constant except kilovoltage, record dose rate at 10 kvp intervals from 30 to 100 kvp.

kvp	r/min
30	
40	
50	
60	
70	
80	
90	
100	

Graph on page 4.

3. Effect of filtration on dose rate, and effect of kvp on penetration:

(a) Operate the tube at 50 kvp and 4 ma. Record the dose rate at 36" with 0.5 mm increments of aluminum filtration. The roentgen ratemeter should be set on the 3 r/min scale.

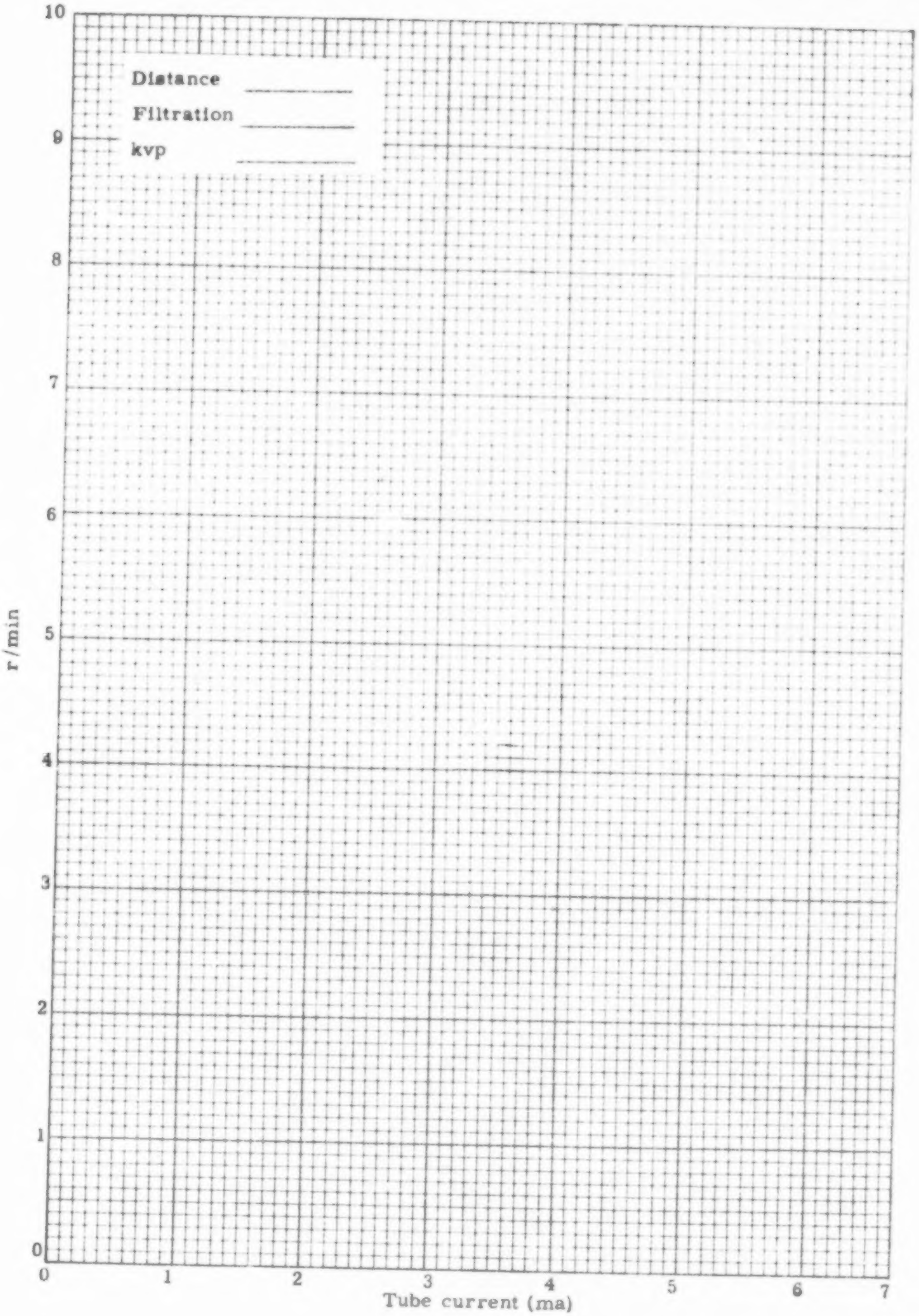
(b) Repeat the above at 100 kvp and 4 ma. The roentgen ratemeter should be set on the 10 r/min scale.

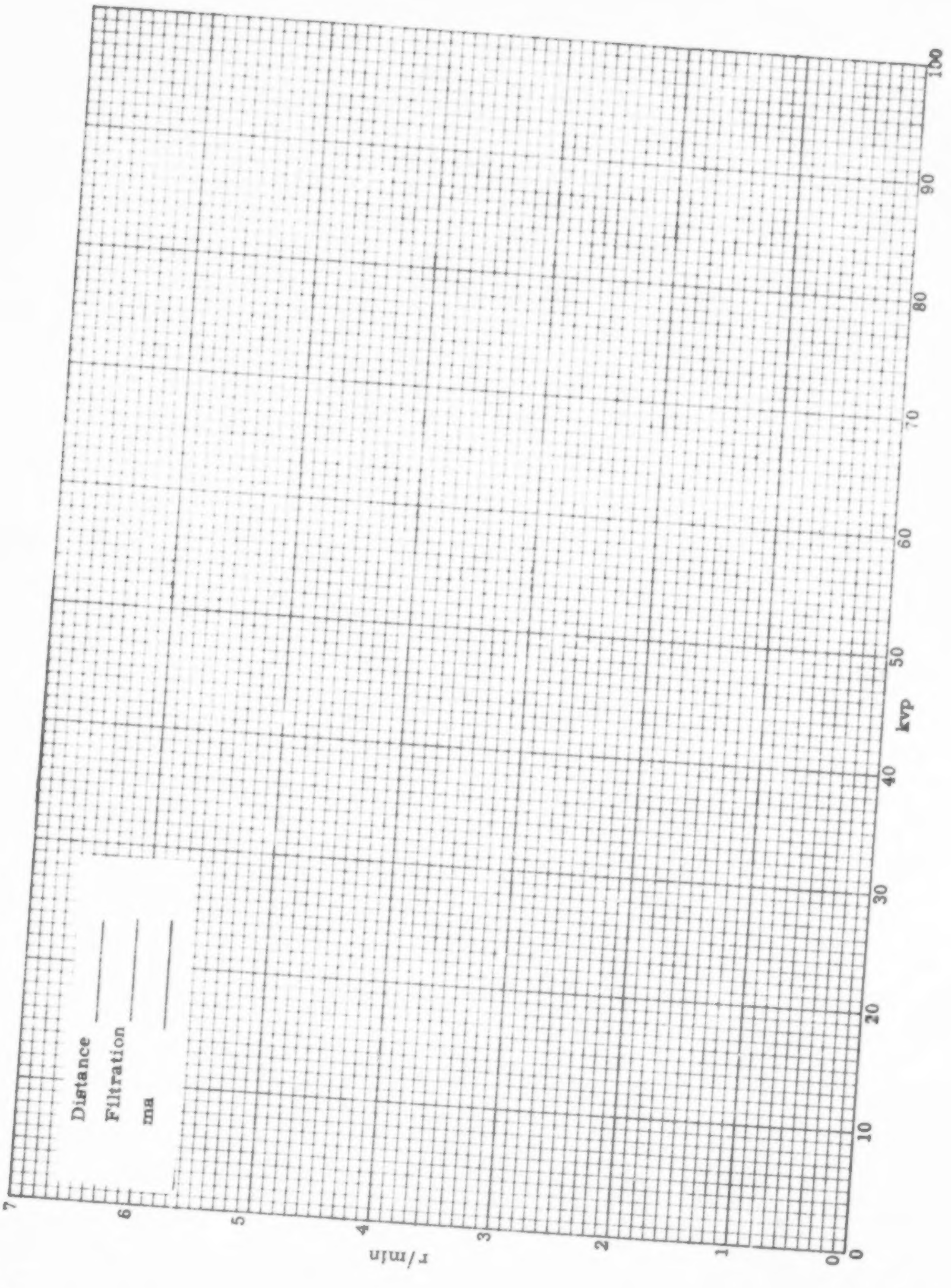
mm Al	50 kvp r/min	100 kvp r/min
0		
0.5		
1.0		
1.5		
2.0		
2.5		
3.0		
3.5		

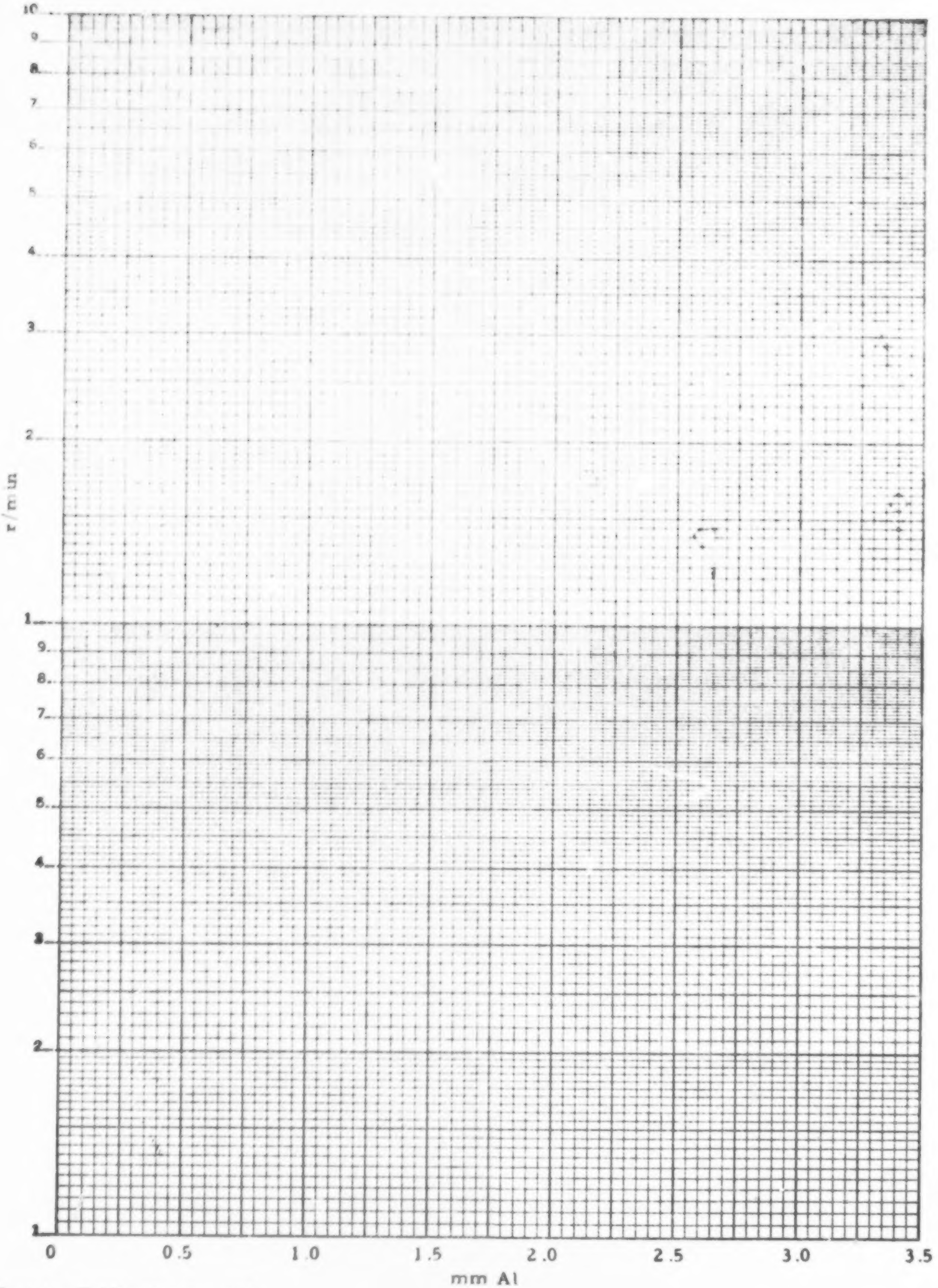
Draw both absorption curves on the same semilogarithmic graph sheet on page 5.

4. Effect of distance on dose rate

Operating under conditions designated by instructor (suggested; 100 kvp, 4 ma, 0.5 mm Al), measure the dose rate at several


Distance ______
Filtration ______
kvp ______
r/min
10
9
8
7
6
5
4
3
2
1
0
0
1
2
3
4
5
6
7
Tube current (ma)


Distance
Filtration
ma
r/min
0
1
2
3
4
5
6
7
kvp
0
10
20
30
40
50
60
70
80
90
100


10
9
8
7
6
5
4
3
2
r/min
1
9
8
7
6
5
4
3
2
1
0
0.5
1.0
1.5
2.0
2.5
3.0
3.5
mm Al

measured distances. Assuming the first reading to be correct, calculate the theoretical dose rates at the other points.

kvp _______ ma ______ filtration _______

Distance (feet)	r/min read	r/min calc

Plot both read and calculated dose rates vs. distance on logarithmic paper on page 7.

5. Reduction of patient dose in diagnostic radiology

Coning: The instructor will demonstrate the effect of cones and diaphragms, by means of fluorescent materials in a darkened room.

Technique:

(a) Direct the beam horizontally toward the phantom*. Operating at 65 kvp, 5 ma, no filtration, target-to-skin distance; 36" measure exit dose, using low range probe on roentgen ratemeter. Record exit dose in mr/min.

(b) Repeat at 90 kvp, but reduce current until exit dose is same as before. Record milliamperes.

(c) Add 2.5 mm Al filtration. Repeat exposure at 90 kvp, but increase current until dose is again the same as before. Record milliamperes.

(d) Place high range probe on roentgen ratemeter. Now measure and record entrance doses for each set of conditions recorded in (a), (b), and (c).

Technique				exit dose mr/min	entrance dose r/min
Step	kvp	ma	mm Al		
a	65	5	0		
b	90		0		
c	90		2.5		

6. Scattered Radiation

Using the technique described in 5 (a), but with no cone, irradiate the phantom. Using ionization chamber survey meters, measure the stray radiation at each of the positions indicated on the sketch. With a course participant and instrument at each position, all readings can be made simultaneously.

Repeat above, but with long cone and technique described in 5 (c).

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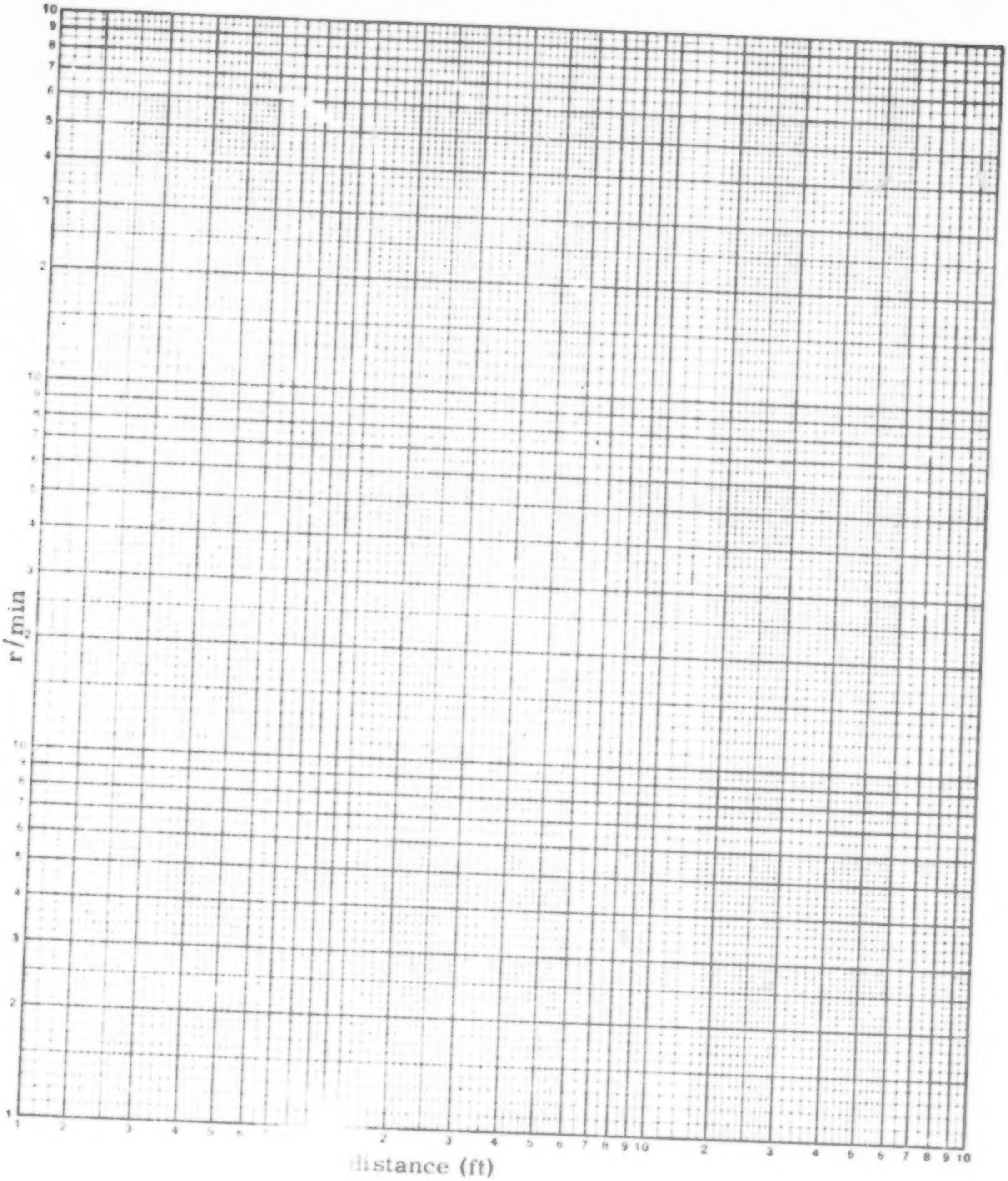
*A phantom is any material or object having approximately the same density, atomic number, and thickness as the human body, and is used to simulate the human body in x-ray experiments.


r/min
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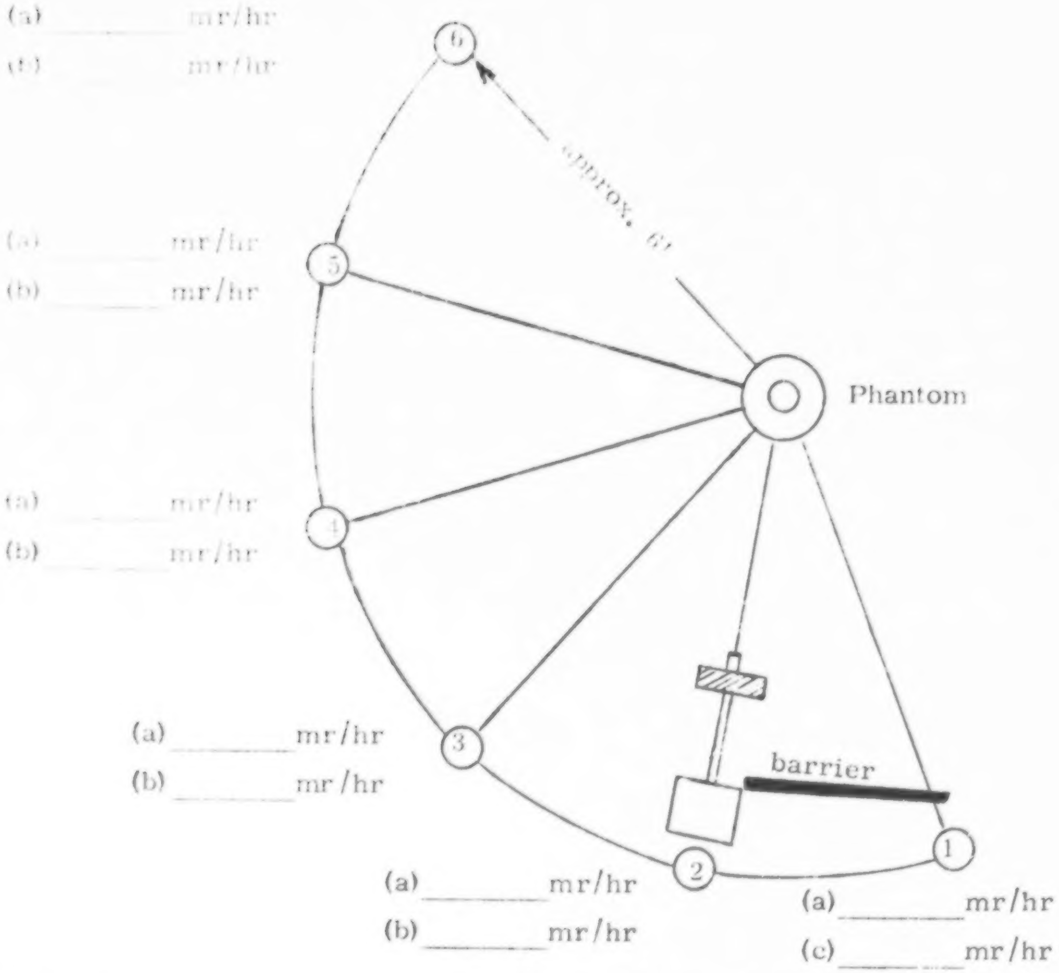
9
8
7
6
5
4
3
2
1
1 2 3 4 5 6 7 2 3 4 5 6 7 8 9 10 2 3 4 5 6 7 8 9 10
distance (ft)


(a) ________ mr/hr
(b) ________ mr/hr
6
approx. 6'
(a) ________ mr/hr
(b) ________ mr/hr
5
Phantom
(a) ________ mr/hr
(b) ________ mr/hr
4
(a) ________ mr/hr
(b) ________ mr/hr
3
barrier
2
1
(a) ________ mr/hr
(b) ________ mr/hr
(a) ________ mr/hr
(c) ________ mr/hr

QUESTIONS

1. From the data obtained in part 3, determine (a) the HVL of a 50 kvp beam of x-rays filtered by 1 mm Al plus inherent filtration; (b) the HVL of a 100 kvp beam with the same filtration.

2. What factors would tend to cause deviations from the inverse square law?

3. If biological effect is roughly proportional to entrance dose and photographic effect to exit dose, which technique used in part 5 is most desirable?

4. Explain how reduction of patient dose reduces dose to operating personnel.

Shielding Properties of Common Materials

OBJECT:

To study the relative value of various materials as shielding for gamma radiation and to determine the half-value layer and linear attenuation coefficient for each, using a collimated gamma source.

INTRODUCTION:

The attenuation of gamma radiation by a particular shield is a function of the energy of the radiation, nature of the absorbing material and the thickness of the absorbing material. For a monoenergetic narrow beam of parallel gamma rays incident upon a thin absorber, the dose rate or flux density decreases exponentially. (Such conditions are termed "good geometry.") Mathematically this is expressed as:

(1) $I = I_o e^{-\mu x}$

where I = radiation intensity or dose rate after passing through absorber of thickness x

x = thickness of absorber

I_o = radiation intensity or dose rate incident upon the absorber

e = base of natural logarithms

μ = linear absorption coefficient (units reciprocal to those of x)

A useful quantity is the half-value layer, or half-thickness, which is the thickness of absorber required to reduce the intensity to one-half the initial value. The relationship between the linear absorption coefficient, μ, and the half-value layer (half-thickness), $X_{\frac{1}{2}}$, can be derived from equation (1) above by substituting $X_{\frac{1}{2}}$ for x, and $\frac{I_o}{2}$ for I. This can be simplified to give:

(2) $$\mu = \frac{0.693}{X_{\frac{1}{2}}}$$

If the radiation does not meet narrow beam conditions and/or thick absorbers are involved, the equation becomes:

(3) $I = bI_o e^{-\mu x}$

where b is the build-up factor, and takes into account an increase in radiation intensity due to scattered radiation within the absorber. (to a first approximation, values for b can be taken to equal $1 + \mu x$.)

LIST OF EQUIPMENT:

1 lead collimator
gamma source(s)
1 film badge for each participant
1 ionization chamber survey meter for each group
Several equal thicknesses of each absorbing material of the following kinds:
- concrete
- aluminum
- lead
- iron
- wood

Adjust the survey meter to read zero with no radiation present. Place the ionization chamber in the radiation beam, at a distance from the source such that a meter reading of approximately full scale is attained with no absorbing material intervening. This reading will be I_0 and should remain the same provided the ionization chamber is kept in the same position throughout the exercise. The instrument should be checked for zero drift at intervals throughout the experiment.

Exercise 1:

1. Using the various types of absorbers one at a time, place at least four or five known thicknesses near the source and record the corresponding meter readings and total thicknesses.

2. On semi-logarithmic graph paper, plot each value of I/I_0 versus thickness of the absorber. Plot thickness on the linear scale (abscissa) and plot the value of I/I_0 on the logarithmic scale (ordinate).

3. From the curve, determine and record the half-value layer (half-thickness), $X_{\frac{1}{2}}$, of each absorber.

4. Compute the value of μ for each absorber by use of equation (2). Record your results, compare with the value of μ obtained from the Radiological Health Handbook for the corresponding absorbing material and energy of gamma radiation. Graphs of total mass absorption coefficients, μ/ρ, (where ρ is the density of the absorber) can be found on pp. 144-146 of the Radiological Health Handbook (1960 edition). Values for μ can be found on page 88 (p. 143 for concrete), or from the center line of the nomograph on page 160.

5. List of absorbing materials examined in the order of their shielding effectiveness.

Exercise 2:

1. Keep the ionization chamber and source in the same geometrical locations and place approximately one HVL of steel near the detector and note the respective intensity reading.

2. Compare the reading of Exercise 1 with that obtained in Exercise 2 for the corresponding absorber thickness. What could account for an observed difference?

Data Sheet

Name ________________________________ Date ______________

Source ______________________ Energy of Radiation ______________

Exercise 1:

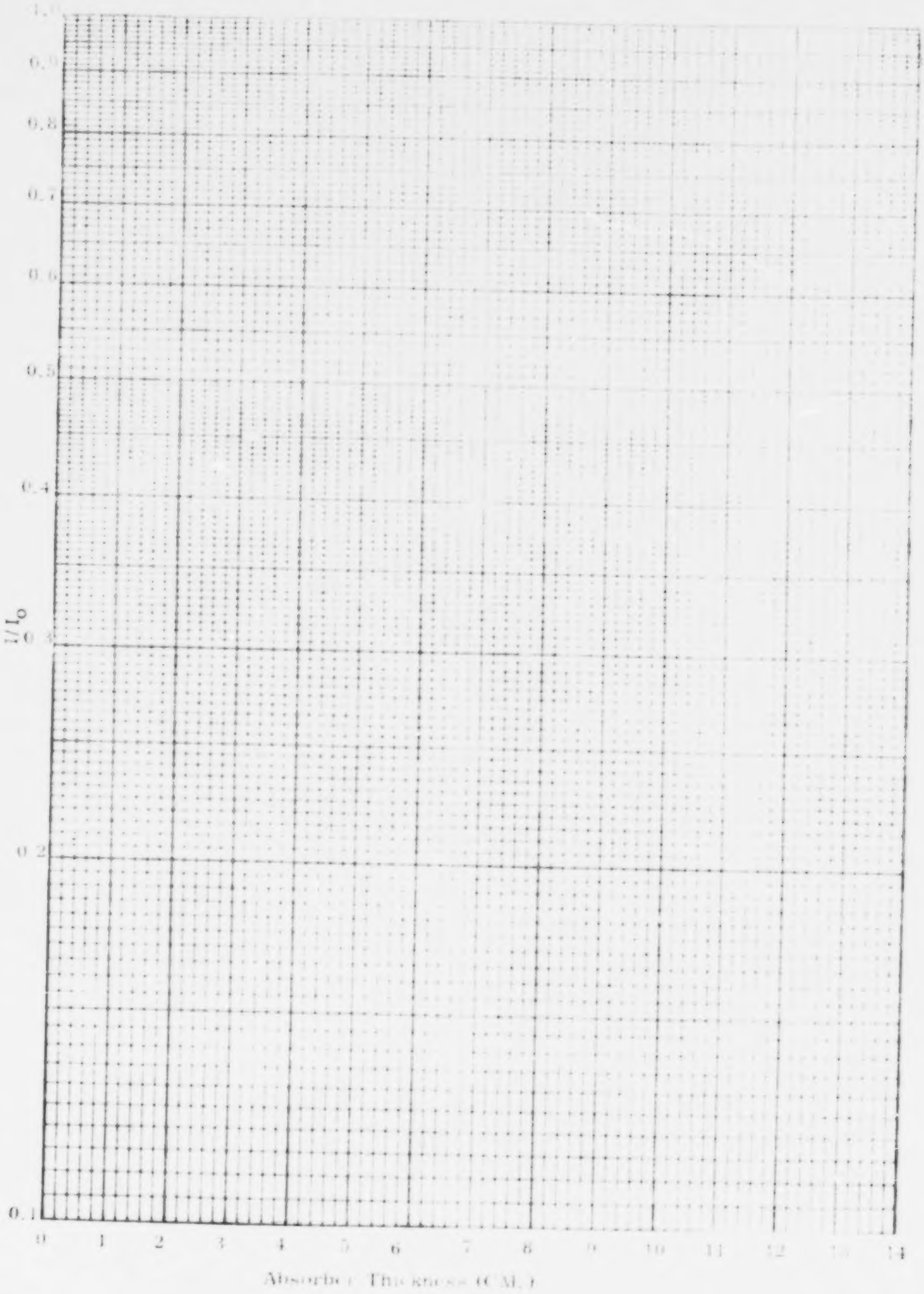
Absorber	Thickness	Intensity			$X_{\frac{1}{2}}$	Absorption Coefficient (μ)	
		I_o	I	I/I_o	Exp.[a]	Experimental[b]	Published[c]

[a] Experimental value from graph

[b] Experimental value: $\mu = \frac{.693}{X_{\frac{1}{2}}}$

[c] Value obtained from Radiological Health Handbook: $\mu = (p)(\mu/p)$

Exercise 2


1.0
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0.7
0.6
0.5
0.4
I/I_0 0.3
0.2
0.1
0
1
2
3
4
5
6
7
8
9
10
11
12
13
14
Absorber Thickness (CM.)

Data Sheet (Cont'd)

Exercise 1: (Cont'd)

List of absorbers according to shielding effectiveness:

Exercise 2:

Absorber Location	I_0	I	I/I_0
Near Source			
Near Detector			

Assay of Radionuclides: Beta Emitters

(Laboratory Experiment)

OBJECT:

To acquaint the student with procedures for the identification of unknown beta-emitting radionuclides.

INTRODUCTION:

For positive identification of a beta-emitting radionuclide, two or more of the following list of its characteristics must usually be known:

1. its half life
2. its chemistry
3. the maximum energy of its beta emissions

LIST OF EQUIPMENT:

Unknown Beta Emitting Samples
Laboratory Counters
Calibrated Aluminum Absorbers
Data Sheet and Graph Paper

PROCEDURE:

Emphasis in this laboratory session will be placed upon determination of maximum beta energy. For this purpose, the following procedure is suggested:

(1) Set up the sample to give a convenient counting rate in a conventional laboratory GM counter.

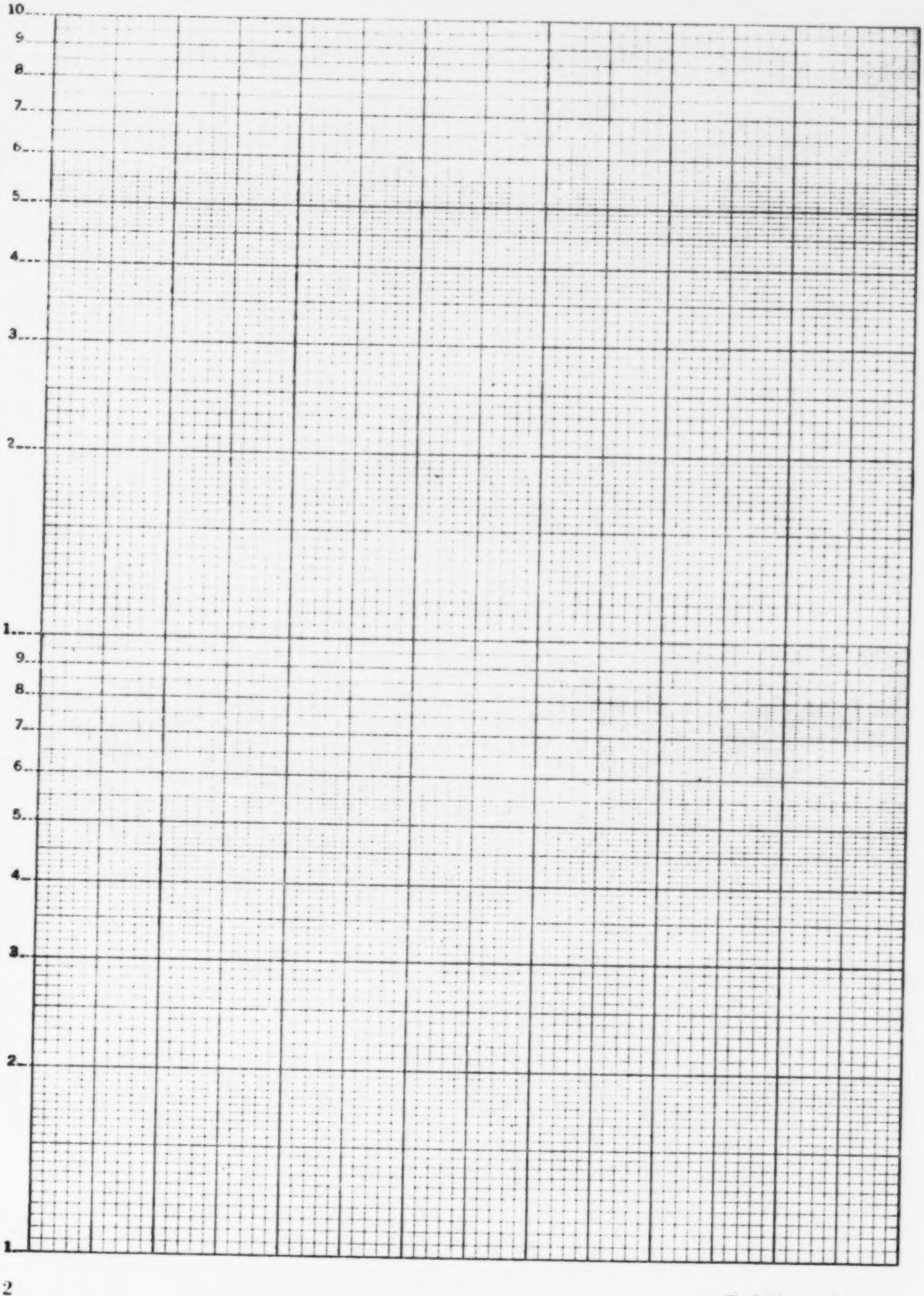
(2) Insert aluminum absorbers of known density-thickness (mg/cm^2) between sample and GM window.

(3) Plot (c/m) vs. (mg/cm^2) on semi-log paper with (c/m) on the log axis. Bear in mind that even with no added aluminum absorber in place, the GM tube window and intervening air space contribute about 5 mg/cm^2 absorption.

(4) From this plot, determine the aluminum half-thickness, i.e., the thickness of aluminum required to reduce the count rate by one-half.

(5) Obtain the maximum energy of the beta rays from an empirical plot of energy vs. aluminum half-thickness for many different emitters; such a plot is reproduced on the following page.

(6) On the basis of the determined value of E_{max} and half life data (to be supplied by laboratory instructor), attempt to identify the unknown radionuclide. A list of known radionuclides arranged according to maximum beta energy is given in the Radiological Health Handbook.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2

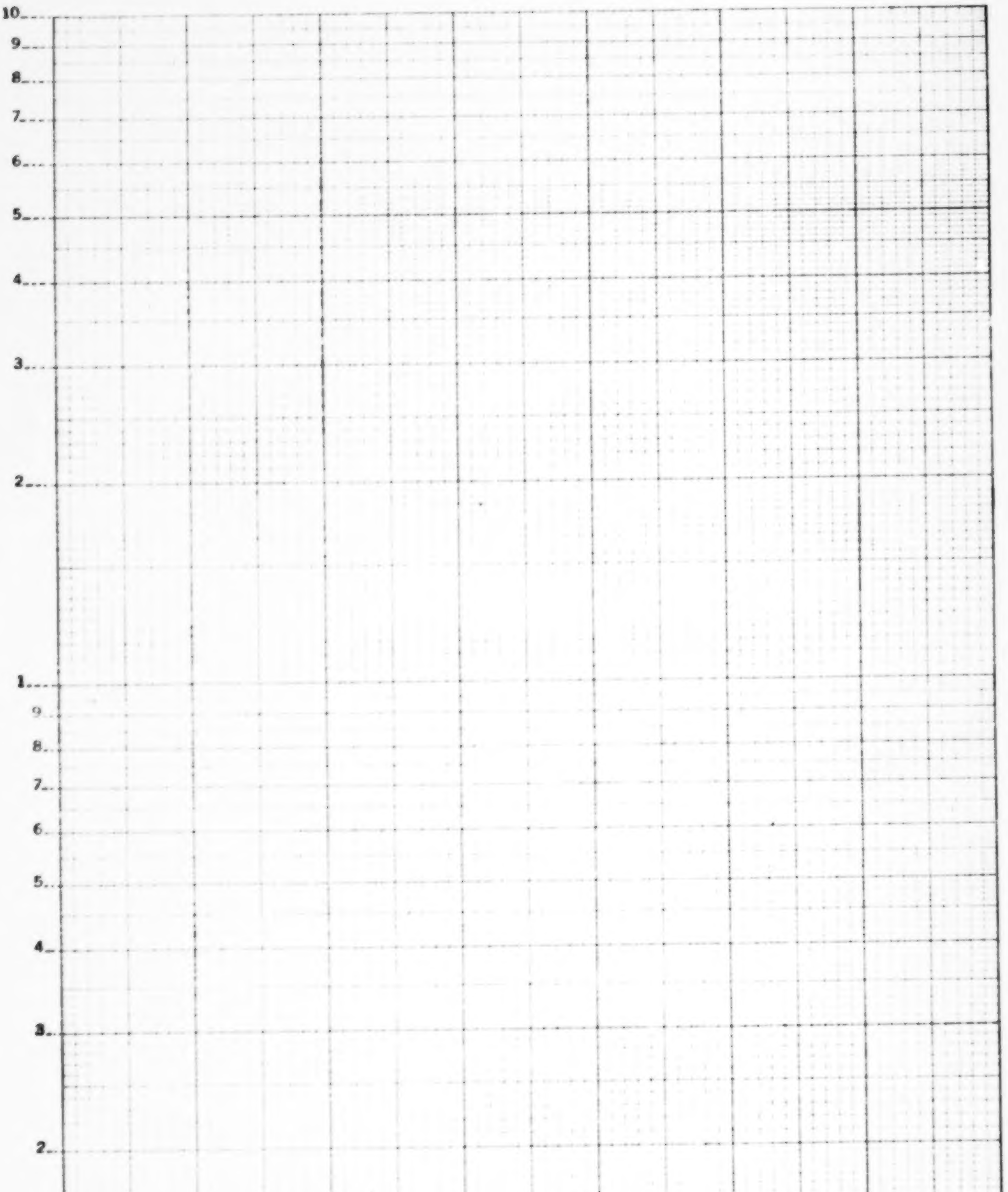
Laboratory Counting Data Sheet

Operator's Name ______________________________ Date ____________

Counter Type ______________________ Serial No.: Counter ________ Shield ____

Operating Voltage ______________________

Sample	Time of Count	Total Count	Gross c/m	Background	Net c/m	mg/cm^2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1

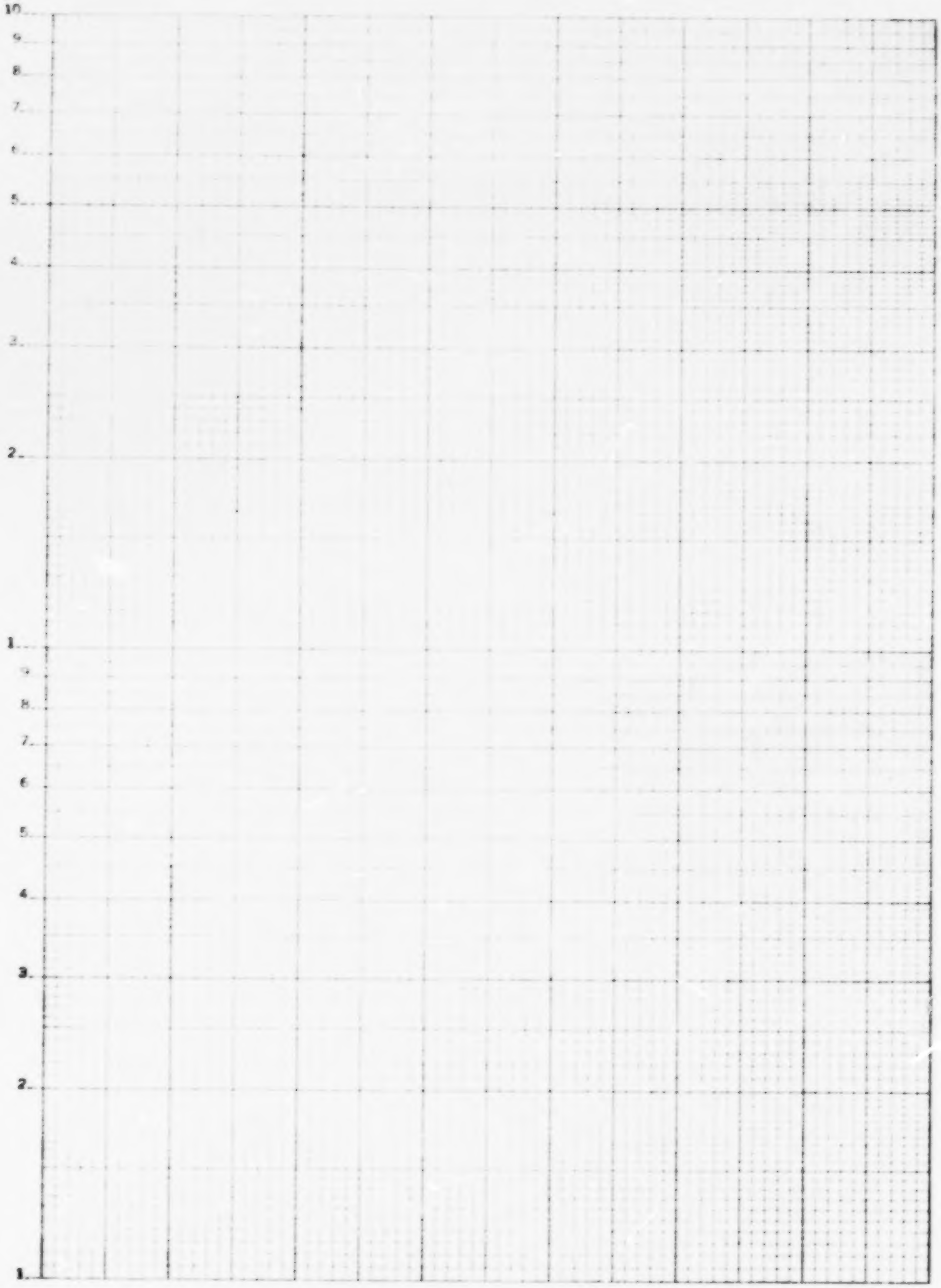
Laboratory Counting Data Sheet

Operator's Name ______________________ Date ________

Counter Type ______________________ Serial No.: Counter ________ Shield ________

Operating Voltage ______________________

Sample	Time of Count	Total Count	Gross c/m	Background	Net c/m	mg/cm^2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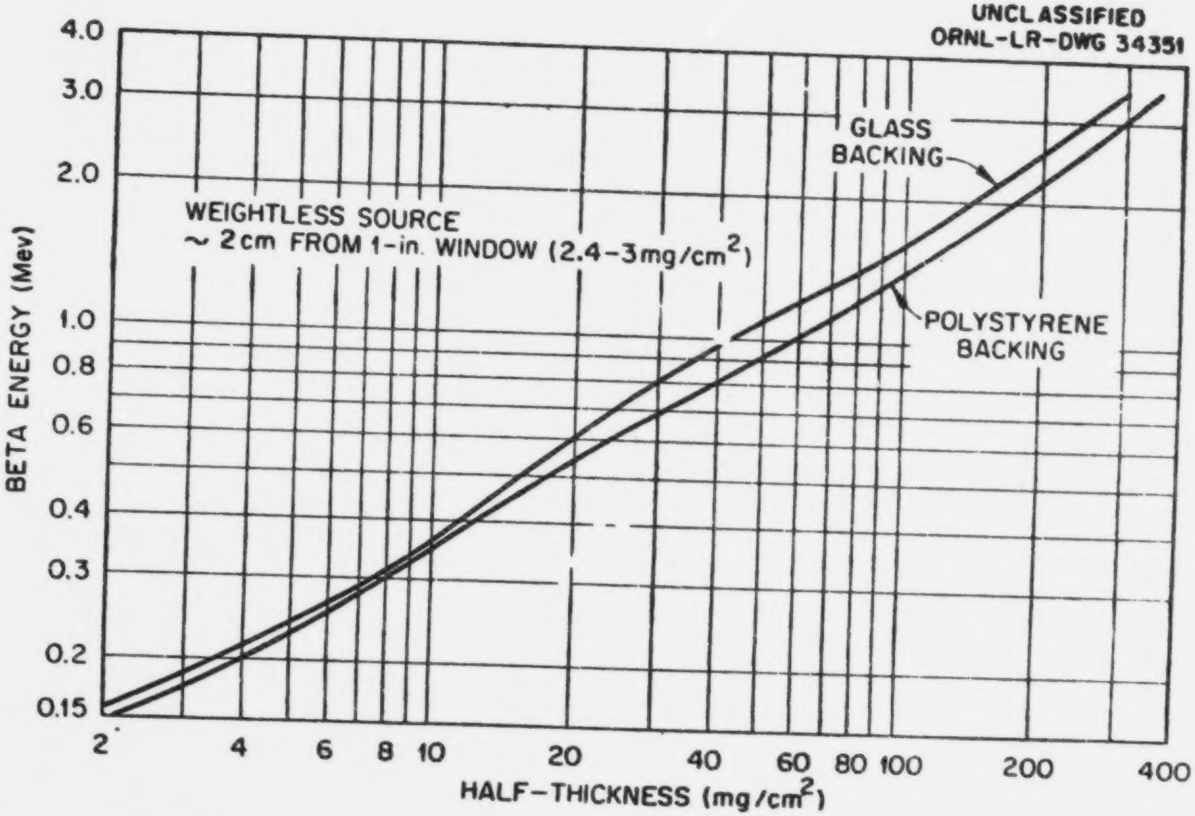
Laboratory Counting Data Sheet

Operator's Name ______________________ Date ____________

Counter Type ______________ Serial No.: Counter __________ Shield ______

Operating Voltage ______________________

Sample	Time of Count	Total Count	Gross c/m	Background	Net c/m	mg/cm^2



Beta Radiation Initial Half-Thickness in Aluminum vs. Maximum Energy.

Assay of Radionuclides: Gamma Emitters

OBJECT:

To acquaint the student with procedures for the identification and quantitative determination of unknown gamma-emitting radionuclides.

INTRODUCTION:

A. Identification

For positive identification of a gamma-emitting radionuclide, two or more of the following list of its characteristics must usually be known:

1. Number and energies of its gamma emissions
2. Its half-life
3. Its chemistry
4. The maximum energy of its beta emissions (if beta emission accompanies its decay).

In general, an attempt is always made to determine the first two characteristic properties of a given unknown radionuclide, with its chemistry serving as a confirmation where needed. It is only in rare cases that the determination of all four of the listed characteristics is necessary for identification.

Gamma energies are most easily determined through use of a gamma spectrometer. Half-life values can be determined either from periodically obtained spectral data or from a plot of gross gamma decay. The chemistry of a particular radionuclide can be determined through application of the usual qualitative tests. Techniques for determination of maximum beta energy usually involve measurements of the decrease in counting rate as increasing amounts of absorber are placed between the sample and a beta detector.

B. Quantitative Determination

Once identification of the radionuclide is complete, several approaches may be applied for obtaining quantitative information. One approach might involve chemical separation of the radioelement in pure form, followed by determination of its activity on a gross gamma counter through application of appropriate factors for chemical separation and counting efficiency. A second approach is to compare photopeak areas for the identified radionuclide with photopeak areas for standard sources of the same nuclide, similarly counted on the same spectrometer.

LIST OF EQUIPMENT

Single-Channel Gamma Scintillation Spectrometer
Multi-Channel Gamma Scintillation Spectrometer
Selected Standard Gamma-Emitting Sources
Samples containing one or more unknown gamma-emitting radionuclides
Chemical Separation Equipment
Personnel Monitoring Devices and Protective Clothing
Punch Card Set Cataloging Decay and Half-life Characteristics of Radionuclides
Data Sheet and Graph Paper

PROCEDURE:

During the course of the laboratory exercise, the students will have an opportunity to operate spectrometers of both the single and multi-channel type. The procedure to be followed in the operation of both type instruments is similar.

Calibration:

In order to determine the energies of the unknown gamma-emitting sample, each spectrometer must be calibrated with sources of gamma activities of known energies. Inasmuch as the unknown samples will emit gamma photons possessing a wide range of energies, it will be necessary to calibrate each instrument for several different gain settings.

Unknown Sample:

1. CHEMICAL SEPARATION

If initial studies show the sample contains too many radionuclides for accurate interpretation of spectral data, chemical separation techniques may have to be employed. The laboratory instructor will suggest appropriate procedures.

2. SPECTRAL DETERMINATION

Periodic determinations should be made of the spectral distribution for the unknown sample through suitable counting operations on either the single- or multi-channel spectrometers.

3. HALF-LIFE DETERMINATION

The half-life of the unknown sample should be determined through periodic gross gamma counting determinations or through comparison of photopeak heights on the several spectral curves obtained at intervals.

4. IDENTIFICATION

Taking into account the knowledge thus obtained concerning the gamma energies, half-life, and chemistry of the unknown sample, the student should identify as completely as possible all radionuclides in the unknown.

5. QUANTITATIVE DETERMINATION

Taking into account data regarding efficiencies for chemical separation and counting, the student should estimate the quantity of each radionuclide identified.

QUESTIONS:

1. What governs the number of radionuclides which can be spectrally identified in one sample?

2. Discuss the relative merits of the single- and multi-channel units.

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2. Reynolds, S. A., "Analytical Radiochemistry", Record of Chemical Progress, Volume 16, Number 2, pages 99-119, 1955.
3. Connally, R. E., "Instrumental Methods of Gamma-Ray Spectrometry", Analytical Chemistry, Volume 28, Number 12, pages 1847-1853, December, 1956.
4. Bell, P. R., "The Scintillation Method", in "Beta and Gamma Ray Spectroscopy", edited by K. Siegbahn, Interscience, New York, 1955.
5. Moeller, D. W., "Radionuclides in Reactor Cooling Water - Identification, Sources and Control", Oak Ridge National Laboratory Report ORNL-2311, June 12, 1957.

Operator ____________________
Date ______________ Time ______
Energy Span ________________
Calibration ________________
Crystal ____________________
Sample _____________________

	Total	c/m		Total	c/m
1			[illegible]		
2			[illegible]		
[illegible]			[illegible]		
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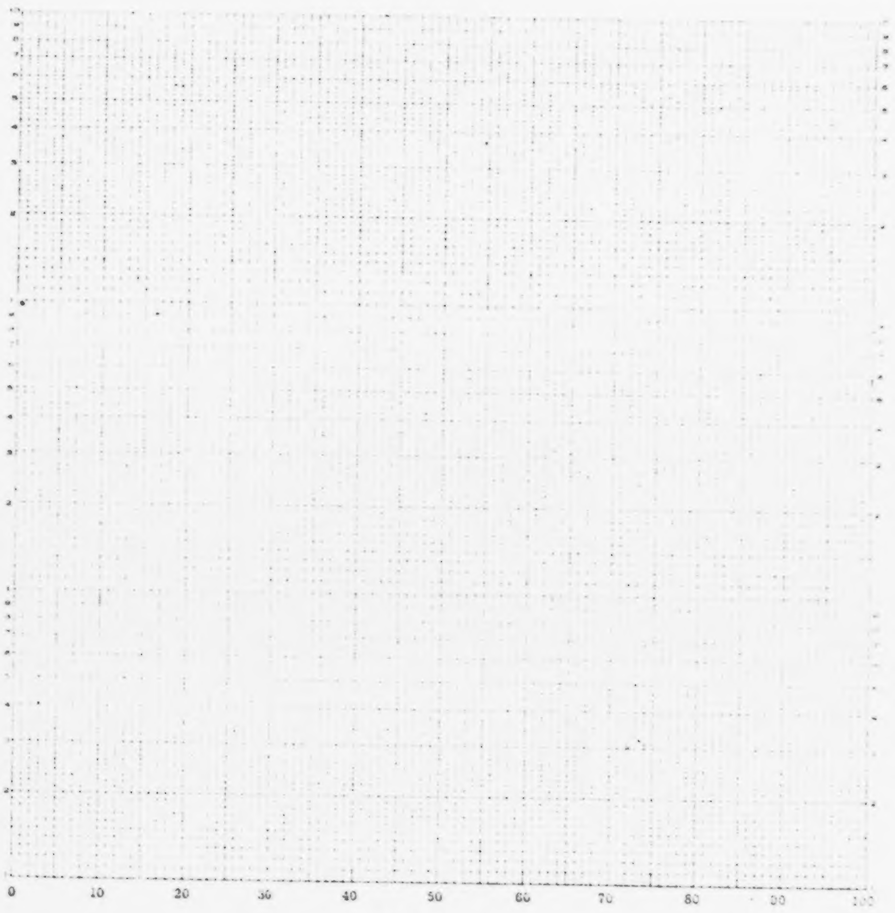

Training Publication No. 2-3b

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Energy Span
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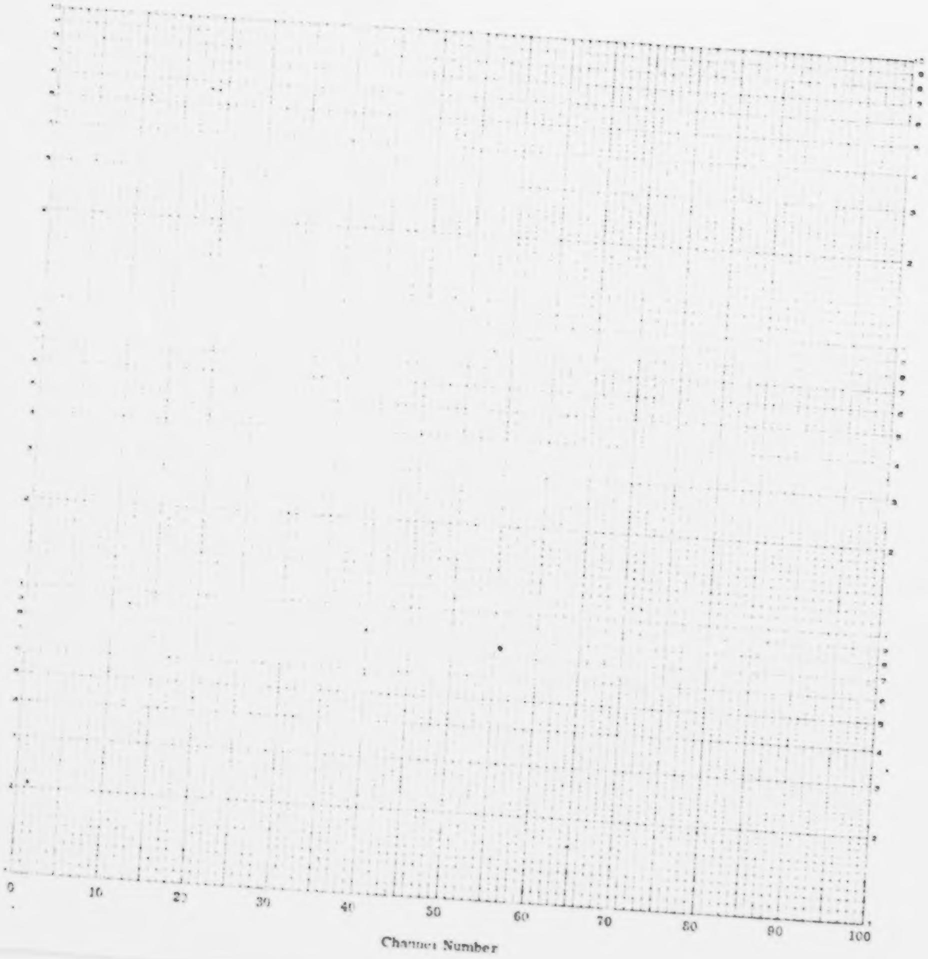
	Total	c/m		Total	c/m



Radionuclides Arranged by Gamma-Ray Energy

By GILBERT W. SMITH and DONALD R. FARMELO
Curtiss-Wright Corp., Quehanna, Pennsylvania

THE TABLE presented here lists radionuclides in order of the energies of their emitted photons. The listing covers the literature to October, 1956, as given by the Table of Isotopes (*1*) and the annual cumulations of nuclear data in Nuclear Science Abstracts.

The following criteria have been used in selecting the items:

- Nuclides derived from n,γ reactions and fission products having >~0.5% yield are included.
- Half-lives or precursor half-lives are a few minutes or longer.
- Gammas <10% of the total activity of any nuclide are omitted.

For use in identifying nuclides by gamma-scintillation spectrometry, the list includes other data along with photopeak energy. The list is also useful in finding photon energies for calibration purposes.

Photopeaks must not be confused with other peaks (*2*): (a) Pair-production peaks appear 0.51 and 1.02 Mev below the photopeak and are especially bothersome if the photopeak occurs above 1.5 Mev. (b) Annihilation radiation appears at 0.51 Mev in positron emitters. (c) Compton collisions that scatter the photon at 180 deg create a peak at the following energy: $E = E_0/(1 + 3.9137E_0)$ with both energies in Mev (*3*). (d) The energy of the K_α X-ray line in kev can be computed from its wave length λ in angstroms as 12.34/λ.

Radiochemical separations may be necessary before using the table if a complex mixture has more than 5 photopeaks. Half-life, detection efficiency, daughter activities, and other properties can be used to resolve situations of this kind.

BIBLIOGRAPHY

1. J. M. Hollander, I. Perlman, G. T. Seaborg, Revs. Mod. Phys. **25**, 469 (1953).
2. P. R. Bell, "The Scintillation Method" in "Beta and Gamma Ray Spectroscopy," K. Siegbahn, ed. (Interscience Publishers, New York, 1955).
3. B. Crasemann, H. Easterday, NUCLEONICS **14**, No. 6, 63 (1956).

Radionuclides by Increasing Gamma-Photon Energy

Photopeak energy (Mev)	*Nuclide*	*Half-life*	*Production cross section* (barns) or fission yield (%)*	*Photon yield per disintegration (%)*	*Daughter*
0.037	Br^{80m}	4.58h	2.9b	100	Br^{80}
0.040	Rh^{103m}	57m	2.9%	100	Rh^{103}(st)†
0.047	Pb^{210}	22y		~100	Bi^{210}
0.049	Br^{80m}	4.58h	2.9b	100	Br^{80}
0.052	Rh^{104m}	4.4m	12b	100	Rh^{104}
0.058	Gd^{159}	18h	4b		Tb^{159}(st)
0.068c‡	Ta^{182}	112d	21b		W^{182}(st)
0.077	Pt^{197}	18h	1.1b	100	Au^{197}(st)
0.080	Ho^{166}	27.3h	60b		Er^{166}
0.081	Xe^{133}	5.27d	6.5%		Cs^{133}(st)
0.084	Tm^{170}– Yb^{170m}	129d	125b		Yb^{170}(st)
0.084	Th^{228}	1.9y		28	Ra^{224}
0.087	Pd^{109}	13.6h	12b		Ag^{109m}
	Lu^{176m}– Hf^{176m}	3.7h– short	40b	100	Hf^{176}(st)
[illegible]	Te^{127m}	110d	0.09b		Te^{127}
[illegible]	Nd^{147}	11.3d	1.8b,2.6%		Pm^{147}
0.093	Th^{234}	24.1d		20	Pa^{234}
0.094	U^{235}	7.1×10^{8}y			Th^{231}
0.100	Sm^{153}	47h	140b		Eu^{153}
0.103	Gd^{153}	236d	<125b		Eu^{153}(st)
0.105	Pa^{233}	27.4d	~50b		U^{233}
0.105	Sm^{155}	2.4m	5.5b	100	Eu^{155}
0.106	Te^{129m}	33.5d	.015b,0.34%	100	Te^{129}
0.110	Te^{125m}	58d	5b	100	Te^{125}(st)
0.112	Lu^{177}	6.8d	4×10^{3}b		Hf^{177}(st)

Photopeak energy (Mev)	*Nuclide*	*Half-life*	*Production cross section* (barns) or fission yield (%)*	*Photon yield per disintegration (%)*	*Daughter*
0.113c	Er^{171}	7.5h	9b		Tm^{171}
0.123c	Eu^{154}	16y	420b		Gd^{154}(st)
0.128	Cs^{134m}	3.2h	0.016b	100	Cs^{134}
0.129	Os^{191}	16.0d	8b	100	Ir^{191}(st)
0.132	Hf^{181}	46d	10b		Ta^{181}(st)
0.134	Ce^{144}	290d	6.0%	30	Pr^{144}
0.136c	Se^{75}	127d	26b		As^{75}
0.140	Tc^{99m}	6.04h	5.8%	99	Tc^{99}
0.142	Ce^{141}	32.8d	0.3b,5.7%	67	Pr^{141}(st)
0.147	Te^{131}	24.8m	0.22b,2.9%	55	I^{131}
0.150	Cd^{111m}	49m	0.2b	100	Cd^{111}
0.150	Kr^{85m}	4.36h	0.1b,1.1%		Kr^{85}
0.155	Re^{188}	16.9h	70b		Os^{188}(st)
0.16	Ba^{140}	12.8d	4.0b,6.3%	60	La^{140}
0.16	Xe^{131m}	12.0d	2.9%	§	Xe^{131}(st)
0.162	Ba^{139}	85.0m	0.5b,6.2%	26	La^{139}
0.176	Se^{83}	25m	0.004b		Br^{83}
0.177	Te^{131m}	30h	0.42%		Te^{131}
0.180	Ta^{182m}	16.5m	0.03b	100	Ta^{182}
0.184	U^{235}	7.1×10^{8}y			Th^{231}
0.188	Ra^{226}	1622y		6	Ru^{222}
0.190	In^{114m}	49d	56b	100	Cd^{114}
0.191	Mo^{101}	14.6m	0.20b,5.4%		Tc^{101}
0.206	Lu^{177}	6.8d	4×10^{3}b		Hf^{177}(st)
0.216c	Hf^{180m}	5.5h	75b	100	Hf^{180}
0.231	Te^{132}	77.7h	4.4%	40	I^{132}
0.234	Xe^{133m}	2.3d	6.5%	100	Xe^{133}

Photo-peak energy (Mev)	Nuclide	Half-life	Production cross section* (barns) or fission yield (%)	Photon yield per disintegration (%)	Daughter†
0.239	Pb^{212}	10.6h		~80	Bi^{212}
0.244c	Eu^{152}	13y	7×10^{3}b		Gd^{152}(st)
0.246	Cd^{111m}	49m	0.2b	100	Cd^{111}(st)
0.250	Xe^{135}	9.13h	5.9%	100	Cs^{135}
0.25	Sm^{155}	24m	5.5b	100	Eu^{155}
0.26c	Ge^{77}	12h	0.2b		As^{77}
0.265	Ge^{75}	82m	0.5b		As^{75}(st)
0.265c	Se^{75}	127d	26b		As^{75}
0.279	Hg^{203}	47d	3.8b	100	Tl^{203}(st)
0.284	I^{131}	8.05d	2.9%		Xe^{131m}§
0.285	Pm^{149}	50h	1.3%		Sm^{149}(st)
0.290	Ce^{143}	33h	1.0b,5.4%		Pr^{143}
0.305	Kr^{85m}	4.36h	0.1b,1.1%		Kr^{85}
0.307	Tc^{101}	14m	5.4%		Ru^{101}(st)
0.31c	Pa^{233}	27.4d	50b		U^{233}
0.316c	Ir^{192}	74.4d	700b		Pt^{192}(st)
0.32	Nd^{147}	11.3d	1.8b,2.6%		Pm^{147}
0.325	Cr^{51}	27d	11b	9	V^{51}(st)
0.326	Sn^{125}	9.5m	0.2b		Sb^{125}
0.33c	Ir^{194}	19h	130b		Pt^{194}(st)
0.335	Cd^{115}- In^{115m}	53h- 4.5h	1.1b		In^{115}(st)
0.35	Bi^{211}	2.16m		16	Tl^{207}
0.36	Gd^{159}	18h	4b		Tb^{159}(st)
0.364	I^{131}	8.05d	2.9%	80	Xe^{131m}§
0.37	Ni^{65}	2.56h	2.6b	15	Cu^{65}(st)
0.37	Se^{83}	25m	0.004b		Br^{83}
0.388	Sr^{87m}	2.8h	1.3b	100	Sr^{87}
0.393	Sn^{113}	118d	1.3b		In^{113m}
0.41c	Kr^{87}	78m	2.7%		Rb^{87}
0.412	Au^{198}	2.69d	96b	99	Hg^{198}
0.44	Zn^{69m}	13.8h	0.1b	100	Zn^{69}
0.444c	Hf^{180m}	5.5h	75b	100	Hf^{180}
0.45	I^{128}	25m	6.3b	7	Xe^{128}(st)
0.468c	Ir^{192}	74.4d	700b		Pt^{192}(st)
0.479	Be^{7}	52.9d		11	Li^{7}(st)
0.48	W^{187}	23.9h	34b		Re^{187}(st)
0.490	La^{140}	40.0h	8.4b		Ce^{140}(st)
0.498	Ru^{103}	39.8d	1.2b,2.9%	99	Rh^{103m}
0.513	Sr^{85}	65d	1.0b	100	Rb^{85m}
0.513	Ru^{106}- Rh^{106}	1y- 30s	0.38%	11	Pd^{106}(st)
0.52	Si^{31}	2.65h	0.11b		P^{31}(st)
0.52	Xe^{135m}	15.6m	6.3%	100	Xe^{135}
0.53	I^{133}	20.9h	6.5%	94	$Xe^{133,133m}$
0.53	Nd^{147}	11.3d	1.8b,2.6%		Pm^{147}
0.537	Ba^{140}	12.8d	6.3%,4.0b	30	La^{140}
0.541	Br^{82}	35.7h	3.5b		Kr^{82}(st)
0.549	As^{76}	26.5h	4.2b	50	Se^{76}(st)
0.55	Sr^{91}	9.7h	5.8%		Y^{91}
0.56c	Ge^{77}	12h	0.2b		As^{77}
0.564	Sb^{122}	2.75d	7b		Te^{122}(st)
0.605	Sb^{124}	60d	2.5b	50	Te^{124}(st)
0.609c	Bi^{214}	19.7m			Po^{214}
0.62c	Br^{82}	35.9h	2.6b		Kr^{82}(st)
0.624	Ru^{106}- Rh^{106}	1y- 30s	0.38%	12	Pd^{106}(st)
0.647	I^{131}	8.05d	2.9%	9	Xe^{131m}§
0.657	Ag^{110m}	270d	2.8b	97	Ag^{110}
0.662	Cs^{137}- Ba^{137m}	33y- 2.6m	5.9%	92	Ba^{137}
0.673	I^{132}	2.26h	600b		Xe^{132}(st)
0.69	W^{187}	23.9h	34b		Re^{187}
0.7	Te^{131}	24.8m	0.22b	45	I^{131}
0.72	Zr^{95}	65d	0.10b,6.3%	99	Nb^{95}
0.726	Ru^{105}	4.5h	0.7b,0.85%	100	Rh^{105m}
0.740	Mo^{99}	68h	0.13b,6.1%	20	Tc^{99}
0.764	Nb^{95}	35d	15b,6.3%	100	Mo^{95}(st)
0.77c	Br^{82}	35.9h	2.6b		Kr^{82}(st)
0.775	Zr^{97}- Nb^{97m}	17.0h- 60s	6.1%	100	Nb^{97}
0.796	Cs^{134}	2.3y	26b	~80	Ba^{134}(st)
0.829	Pb^{211}	36.1m		13	Bi^{211}
0.83c	Ga^{72}	14.3h	4.6b		Ge^{72}(st)
0.835	Mn^{54}	291d		100	Cr^{54}(st)
0.835	Ga^{72}	14.3h	3.4b	100	Ge^{72m}
0.845	Mn^{56}	2.58h	13.4b	100	Fe^{56}(st)
0.88	Sc^{46}	84d	12b	99.5	Ti^{46}(st)
0.91c	Rb^{88}	17.8m	0.14b		Sr^{88}(st)
0.95	Cd^{115}- In^{115m}	53h- 4.5h	1.1b		In^{115}(st)
0.95	Se^{83}	25m	0.004b		Br^{83}
0.96	Mo^{101}	14.6m	0.20b,5.4%		Tc^{101}
0.96c	Tb^{160}	73d	44b		Dy^{160}(st)
1.0c	Dy^{165}	2.32h	2.7×10^{3}b		Ho^{165}(st)
1.08	Rb^{86}	18.7d	0.7b	20	Sr^{86}(st)
1.097	Fe^{59}	46d	0.9b	50	Co^{59}(st)
1.11	Zn^{65}	245d	0.5b	45	Cu^{65}(st)
1.11	Sc^{46}	84d	12b	99.5	Ti^{46}(st)
1.12c	Eu^{154}	16y	4.2×10^{3}b		Gd^{154}(st)
1.12	Ni^{65}	2.56h	2.6b	30	Cu^{65}(st)
1.12c	Ta^{182}	112d	21b		W^{182}(st)
1.172	Co^{60}	5.27y	20b	99	Ni^{60}(st)
1.27c	In^{116m}	54m	150b	75	In^{116}
1.27	I^{135}	6.75h	6.3%		$Xe^{135,135m}$
1.277	Na^{22}	2.60y		100	Ne^{22}(st)
1.29	A^{41}- K^{41m}	109m- short	0.53b	99	K^{41}(st)
1.295	Fe^{59}	45.0d	0.9b	50	Co^{59}(st)
1.332	Co^{60}	5.27y	20b	99	Ni^{60}(st)
1.36	Ho^{166}	27.3h	60b	11	Er^{166m}
1.38	Na^{24}	14.90h	0.56b	100	Mg^{24}(st)
1.4	V^{52}	3.77m	4.5b		Cr^{52}(st)
1.44c	Cs^{138}	32.9m	5.7%	100	Ba^{138}(st)
1.49	Ni^{65}	2.56h	2.6b	15	Cu^{65}(st)
1.53	K^{42}	12.44h	1.0b	20	Ca^{42}(st)
1.57	Cl^{38}	37.3m	0.6b	31	A^{38}(st)
1.60	Cl^{38}	37.3m	0.6b	31	A^{38}(st)
1.60	La^{140}	40.3h	8.4b		Ce^{140}(st)
1.61	Pr^{142}	19.21h	10b		Nd^{142}(st)
1.78	Al^{28}	2.27m	0.23b	100	Si^{28}(st)
1.8	I^{135}	6.75h	6.3%		$Xe^{135,135m}$
1.81	Mn^{56}	2.58h	13.4b	30	Fe^{56}(st)
1.85c	Rb^{88}	17.8m	0.14b		Sr^{88}(st)
2.15	Cl^{38}	37.3m	0.6b	47	A^{38}(st)
2.62	Tl^{208}	3.1m		100	Pb^{208}(st)
2.76	Na^{24}	14.90h	0.56b	100	Mg^{24}(st)
2.8c	Rb^{88}	17.8m	0.14b		Sr^{88}(st)

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* Thermal-neutron cross section of target atom.

† (st) = stable element.

‡ c = complex spectrum.

§ 1% abundant in I^{131}.